Ontario Hydro Submission to Ontario Nuclear Safety Review

August, 1987

Ontario Hydro
700 University Avenue
Toronto, Ontario
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Robert C. Franklin, President

July 29, 1987

Dr. Kenneth Hare
Commissioner
Ontario Nuclear Safety Review
180 Bloor Street West, Suite 303
Toronto, Ontario
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3667

Dear Dr. Hare:

The attached document has been prepared in response to your request of March 24, 1987 for an Ontario Hydro submission on specific nuclear safety-related items of interest to the Ontario Nuclear Safety Review. One hundred and fifty additional copies of this report are being forwarded directly to your offices.

Ontario Hydro has attempted to provide a concise summary of the important information related to nuclear safety. We have also provided a list of references and associated documents for people who want more information on any of the subjects covered in the report.

Should you wish to discuss any of the information or want further clarification on any item, please do not hesitate to call.

We look forward to future discussions with you on these important safety related matters.

Sincerely,

R C Franklin

Robert C. Franklin

Enc.

Ontario Hydro Submission to the Ontario Nuclear Safety Review

ONTARIO HYDRO SUBMISSION TO THE ONTARIO NUCLEAR SAFETY REVIEW

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FOREWORD

Electricity was first generated from nuclear energy in the Province of Ontario 25 years ago. Now large nuclear-electric power stations provide more than half of all the electricity generated within the province. This dramatic growth in the use of nuclear energy has brought substantial benefits to Canada and Ontario, including:

Low Electricity Costs

A current saving to the Ontario electricity customers of at least $1 billion per year, and increasing every year.

Prevention of Environmental Damage

No acid gas releases from this 50% of Ontario electricity generation, which is economically preventing release of about a million tons of acid gas this year, with further reductions in the future, and no combustion products, such as carbon dioxide, which is a long-term concern of environmental scientists.

Jobs and Economic Advantage

Currently about 30,000 direct jobs, many of them high-tech, equivalent to nearly 100,000 jobs overall, and an accumulated saving to date in foreign exchange of more than $7 billion, which is also increasing every year.

Energy Security

A long term supply of economic energy, indigenous to the province.

Spin-Off Benefits

High quality engineering and manufacturing capability; state-of-the-art technology for industrial applications in gauging, analysis and tracing; medical and health uses in diagnosis, cancer therapy and instruments and dressing sterilization; and food preservation and crop improvements.

The rapid growth of nuclear energy in Ontario has also brought concerns about radioactive wastes which are being carefully stored away from man's environment until the development of a safe, permanent disposal system is completed and about potential public risks from the large inventory of radioactive materials present in the fuel at the operating nuclear stations, which is the main subject of this report. These concerns are related to consequences of potential exposures to ionizing radiation resulting from an uncontrolled release. While man and all living organisms have always been exposed to ionizing radiation of natural origin without demonstrable ill-health effects, if a release were to result in large exposures to radiation, the potential exists for adverse effects.

This potential danger from nuclear power stations was recognized at the outset of development of nuclear energy. Extensive experiments and studies were undertaken to determine approaches to prevent releases of radioactivity under sustained high power operating conditions, and also under a wide variety of upset and accident situations. These experiments and knowledge base acquired from building and operating the early experimental reactors and demonstration and prototype power stations in many countries throughout the western world has, through open discussion and careful analysis and assessment, formed the basis of regulations and codes of practice for the design, construction and operation of the current nuclear power stations. While these regulations, practices and designs differ in detail and approach from country to country, they all conform to a fundamental safety goal: to make it highly improbable that a release of radioactive materials, at a level damaging to life, will occur. The Canadian safety regulations for nuclear power stations enforced by the Atomic Energy Control Board are amongst the most stringent, if not the most stringent, in the world.

To ensure a full utilization of the above benefits to Ontario now and in the future, and to make nuclear power acceptably safe, the CANDU nuclear power stations in Ontario have been designed and constructed to the highest quality standards and are operated by highly trained, reliable operators. The inherent safety features of the design, complemented by an operating staff supported by well established safety policies, principles, standards and documented work control procedures provide the necessary confidence in the overall system. Even so, human errors and omissions, equipment failures, material defects and deterioration, natural forces and combinations of these events must be expected to occur, and on occasion do occur. These events are anticipated, carefully assessed, and safely managed by designing-in redundancy, diversity, separation and fail-safe features in plant systems. These features are backed up with powerful safety and containment systems to prevent an uncontrolled release of radioactivity to the public environment.

This approach to public safety is unique to the nuclear power industry and is the reason that the nuclear power industry in Canada and the rest of the western world has established an exemplary record of no radiation fatalities to plant staff or the public since the first power reactor began operating over 25 years ago. Over this period, many equipment failures and human errors and combinations of these events have occurred without public harm and much experience has been gained to strengthen and improve safety performance for the future. With continued dedication to safety we expect this record to continue indefinitely in Ontario.

This report to the Commissioner of the Ontario Nuclear Safety Review summarizes a wealth of information on management and experience of safe nuclear power in Ontario.

Management of Nuclear Safety in Ontario Hydro

1.0 MANAGEMENT OF NUCLEAR SAFETY IN ONTARIO HYDRO

1.1 Responsibility for Safety Management

This section presents an overview of the safety management process associated with the design, construction and operation of Ontario Hydro's nuclear power plants (Reference 1) and describes in a general way the basis of the process, the responsibilities and interaction of the various groups within the company and their interaction with regulatory agencies. Other sections of this report will examine various aspects in more detail.

Management of nuclear safety in Canada is principally the responsibility of the owner/operators of nuclear facilities, with the regulatory authority (the Atomic Energy Control Board) establishing publicly and technically acceptable standards and ensuring that the owner/operator (licencee) meets these standards on a continuing basis. The effect of this approach is that the owner/operators of nuclear facilities in Canada place considerable emphasis on assessment of the risk of nuclear accidents.

Ontario Hydro has been a staunch supporter of the Canadian approach to nuclear safety management since the beginning of the nuclear program in Ontario. Public safety is one of the primary considerations governing the operation of Ontario Hydro and has always been an integral part of Ontario Hydro's corporate values. Our corporate policy on health and safety states:

> "Ontario Hydro is committed to achieving an acceptable standard of health and safety for its employees and the public by effectively managing all risks resulting from or associated with its activities and operations".

The criteria for implementation of the corporate policy indicate that Ontario Hydro shall, as a minimum standard, comply with and meet the intent of all applicable legislation. Additionally, Ontario Hydro shall adequately manage all hazards and exposures in providing for the health and safety of its employees and the public.

The corporate policy is implemented by the responsible Branches.

Specifically in the area of nuclear safety, the Nuclear Integrity Review Committee (NIRC) has been constituted to:

- Ensure, based on experience at existing nuclear stations, adequate reliability, effectiveness and independence of all nuclear systems in Ontario Hydro nuclear stations, and

- Ensure that the concept, arrangement and design of nuclear systems in Ontario Hydro stations provide adequate protection for incidents and accidents.

The Nuclear Integrity Review Committee is comprised of senior representatives from the design, operations, training and health and safety groups at Ontario Hydro who are responsible for implementing the objectives of the corporate policy (see Figure 1-1). AECL CANDU Operations also have representation on NIRC.

Formation of NIRC was in response to the recommendations made in the Final Report on the Safety of Ontario Hydro Nuclear Reactors by the Ontario Legislature Select Committee on Hydro Affairs. Deliberations of NIRC are reported annually on its activities to improve the reliability, effectiveness and independence of nuclear systems, highlighting conclusions of completed projects and problems under study. The NIRC annual report is made available to the public and is tabled in the Legislature Assembly by the Minister responsible for Ontario Hydro.

1.2 Nuclear Safety Management: Goals and Objectives

The three basic objectives of the nuclear safety program are:

(a) Protection of the public
(b) Protection of the staff
(c) Protection of the plant

The first objective is to ensure that the risk to the public from the harmful effects of ionizing radiation from operation of nuclear power plants is acceptably low. This encompasses normal operation of the plant as well as risks which may arise from potential accidents. The second objective covers protection against radiation but also against other toxic substances and industrial hazards. Plant protection is also essential if the economic risk to the company from accident-related costs is to be minimized. These three objectives are closely related.

Production reliability and safety requirements drive in the same direction - leading to minimizing the risks associated with nuclear accidents.

A point worth emphasizing is that none of these objectives can be met absolutely. For each of them, there remains a finite residual risk, as there is with all human endeavours.

This section will concentrate on the first of these objectives - the other two are discussed in Sections 7 and 12.

The underlying principle (References 2,3) which has governed the Atomic Energy Control Board's approach to regulation of nuclear energy developments in Canada is that primary responsibility for achieving high standards of nuclear safety in design, construction, commissioning, operation and decommissioning of nuclear plants rests with the licencee. The AECB does this by setting what they consider to be technically and publicly acceptable standards with respect to public risk, and then requiring the owner/operator to define exactly how this standard will be

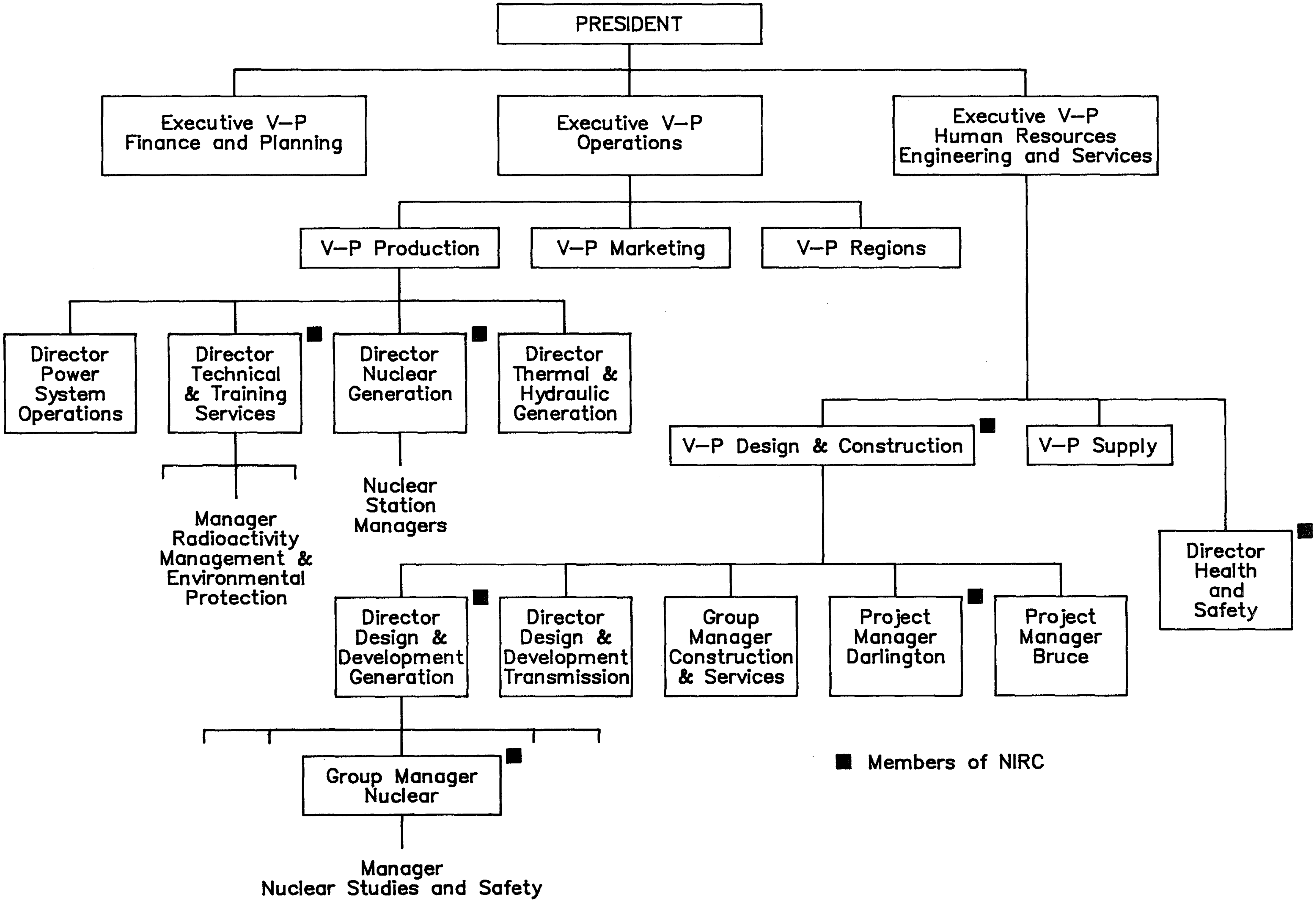


FIGURE 1-1
Organization Structure

met. The role of the Board is one of ensuring that the licencee lives up to its responsibility. The role of the AECB staff is to audit the operating company's performance on behalf of the Board. The role of Ontario Hydro is to ensure that well established safety policies and principles are in place and being followed, that the design and operation of nuclear stations is carried out under clear assignment of responsibility and authority in accordance with written procedures, and that work planning and control is properly reviewed and approved (Reference 4).

This principle of operating company responsibility is generally supported internationally (Reference 5) but the extent to which it is implemented differs widely. Canada and Great Britain, for example, put much more emphasis on this approach than do countries where there tend to be highly detailed rules imposed by the regulatory bodies. The safety culture within the operating company is related directly to that measure of responsibility. In Canada the owners are responsible for proposing, justifying and defending their safety assessments, including the necessary research and development programs to support these assessments. In such an environment, there is considerably more emphasis on understanding the details of, or the consequences arising from, accidents at nuclear plants than in environments where detailed rules exist and R&D is primarily the responsibility of the regulator.

One should draw a clear distinction between the achievement of safety (that is, meeting the previously stated objectives) and the process of nuclear plant licensing (which identifies some of the requirements for meeting objectives 1 and 2). The public is perfectly safe (from radiological hazard) before the plant starts to run, and it cannot be run until an operating licence has been granted to the plant. The licensing authority can provide a checkout of the safety-enhancing characteristics of the facility and of the people involved. In addition, they can help create an environment in which safe practices are likely. Actual achievement of safety, however, is the prime responsibility of the operating company.

1.3 Safety Management

One necessary element for carrying out the corporate safety responsibilities is the establishment of an effective safety management structure. Safety, and good engineering and operational practices are inextricably intertwined, so the safety management structure must of necessity include many parts of the Ontario Hydro organization.

Ontario Hydro has always maintained a high degree of participation in all stages of the design, and complete control of the construction, commissioning and operation of its nuclear stations. At the beginning of the nuclear program Ontario Hydro specified broad project requirements and participated in studies of options to meet these requirements, but left the detailed design of the chosen option and procurement for the nuclear island to Atomic Energy of Canada Limited (AECL). Ontario Hydro has always, however, designed the balance of plant.

As part of the facility design, AECL provided safety assessments which were reviewed by Ontario Hydro and used as the information base for both licensing and operation. With time and the increasing number of facilities being built, Ontario Hydro's participation in design increased to the point where, with Darlington, Ontario Hydro have designed all but the reactor core, shutdown and regulating systems. Safety assessments were carried out entirely within the Ontario Hydro organization.

This highly centralized way of building nuclear facilities, has had the benefit of enabling Ontario Hydro to implement a highly effective safety management process. Those parts of the current organization which are directly involved in the safety management process are highlighted in Figure 1-1.

The following subsections describe the responsibilities of the various groups and their roles in the design and operational phases of a nuclear project.

1.3.1 Safety Management - Design Phase

The Design and Construction Branch, shown in Figure 1-1, is responsible for engineering in Ontario Hydro. On nuclear projects, Atomic Energy of Canada Ltd. acts as consultant on the nuclear steam supply portion, with the scope of their work varying between projects. When a new major project has been committed, a dedicated project team is formed from Design and Development - Generation Division staff under a Project Manager who reports to the Vice-President of the branch. The project manager designates a Manager of Engineering and Manager of Construction and this team is responsible for executing the detailed design and construction of the project.

Within the Design and Development - Generation Division the Nuclear Studies and Safety Department (NSSD) holds the responsibility for nuclear safety. This project responsibility remains with NSSD at all stages in the life-cycle of a major project. The reason for this is to maintain a strong central influence over safety matters.

As shown in Figure 1-2, NSSD provides a key role in the interpretation of regulatory requirements, and the development of the Nuclear Safety Design Guides and the safety system functional requirements. In addition, the department provides the conceptual design of the safety systems, provides the safety analyses and interacts with the regulatory authority on all safety issues. To provide the necessary support for the safety analyses, the Department also administers a large safety R&D program within Canada, and actively contributes to large scale international safety R&D programs. Centralization of these activities within one department produces a speedy and coordinated response to safety issues and provides a continuity and uniformity between the various projects.

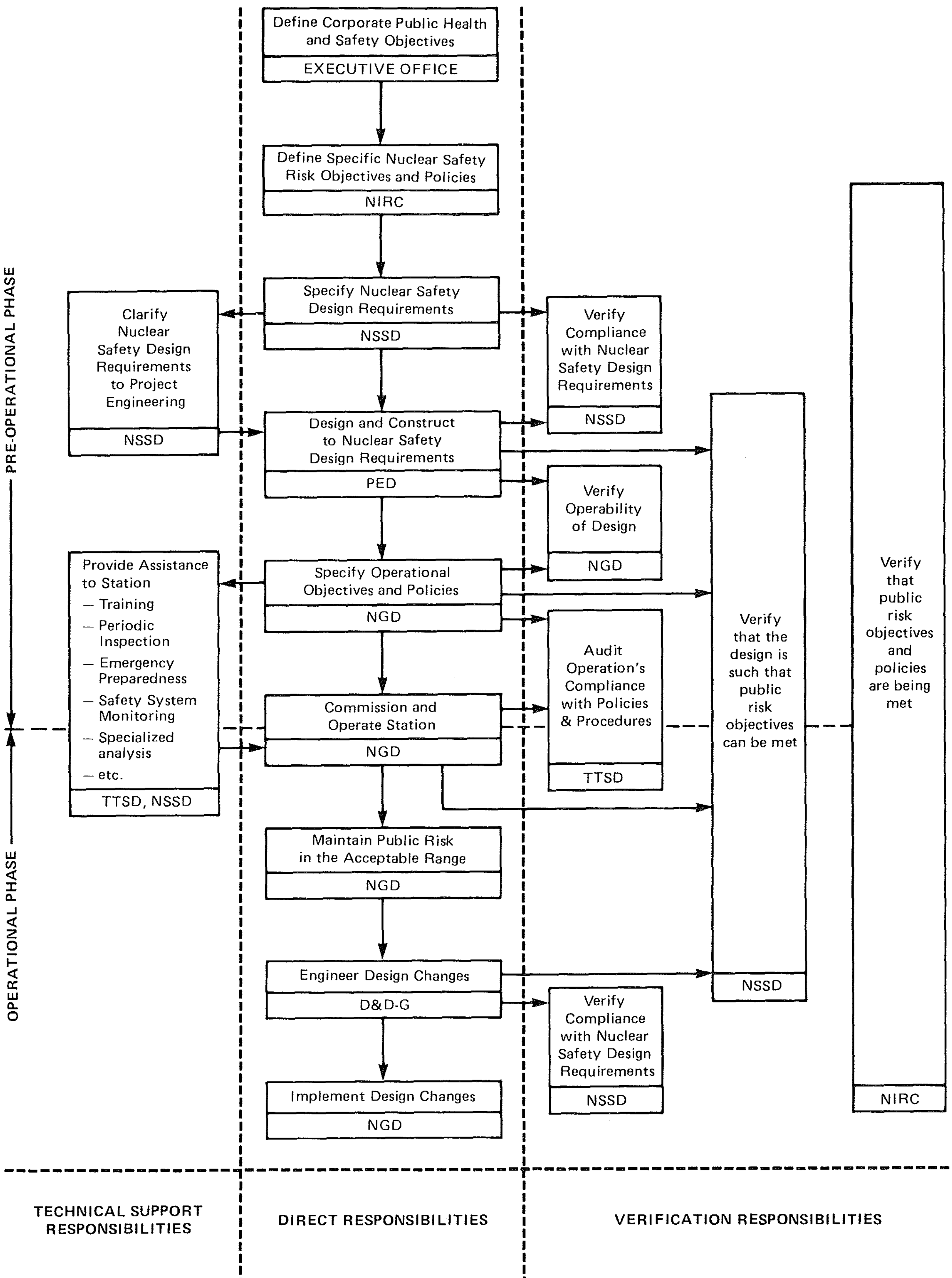


FIGURE 1-2
Public Risk Management Framework and Responsibilities

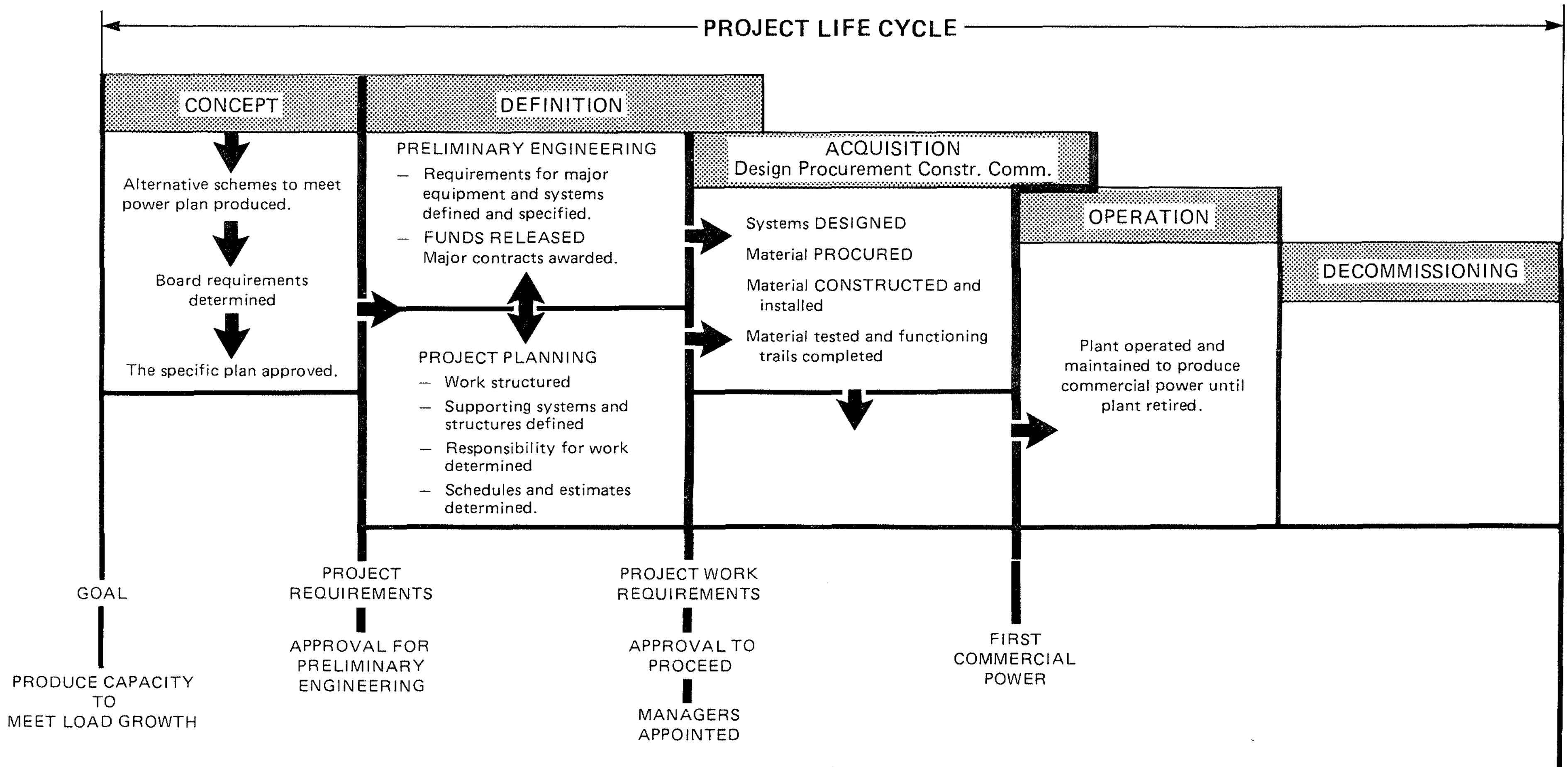
Figure 1-3 shows the phases during the life-cycle of a major project. During the conceptual phase of the project the general characteristics of the design and the site, and the overall plan for the project are established. The AECB is notified of the intent to build a generating station at this stage.

The first important safety management step occurs early at the definition stage. This involves reaching agreement with the regulatory body on the basic ground rules for licensing the plant. These rules clearly identify principles - not details - to be followed throughout the project. Any subsequent disagreements between regulatory staff and the owners/designers about the adequacy of the system design, or adequacy of derived design targets, are resolved with the Board based upon the intent of the guiding principles.

Having established the ground rules the owner can then issue what are called the "Nuclear Safety Design Guides". These documents spell out to the system designers the applicable safety design philosophy and the safety-related requirements to be followed during detailed design. For Darlington NGS A nine Nuclear Safety Design Guides were issued covering the range of topics shown in Table 1-1. Development of these guides can be an iterative process (as shown by the revision numbers in Table 1-1) involving comment from designers, safety analysts and the AECB staff.

The responsibility for approval of these guides rests with the owner, since this is the owners' way of stating how they intend to achieve adequate public protection and adhere to the principles established by the regulatory authority. Information from these guides provide input to the System Design Requirements.

Throughout the design phase of the project the process of verification is implemented by design reviews which are held for selected systems - including the most important safety-related systems. The procedure, formally documented in the Quality Engineering Manual, is illustrated in Figure 1-4. The chairman of the Design Review committee assembles a review team of experienced designers, not involved in the design of the system being reviewed. Questions generated by this team are submitted to the system designers and a review meeting is held to discuss the responses. Following this meeting a Design Review Report is submitted to

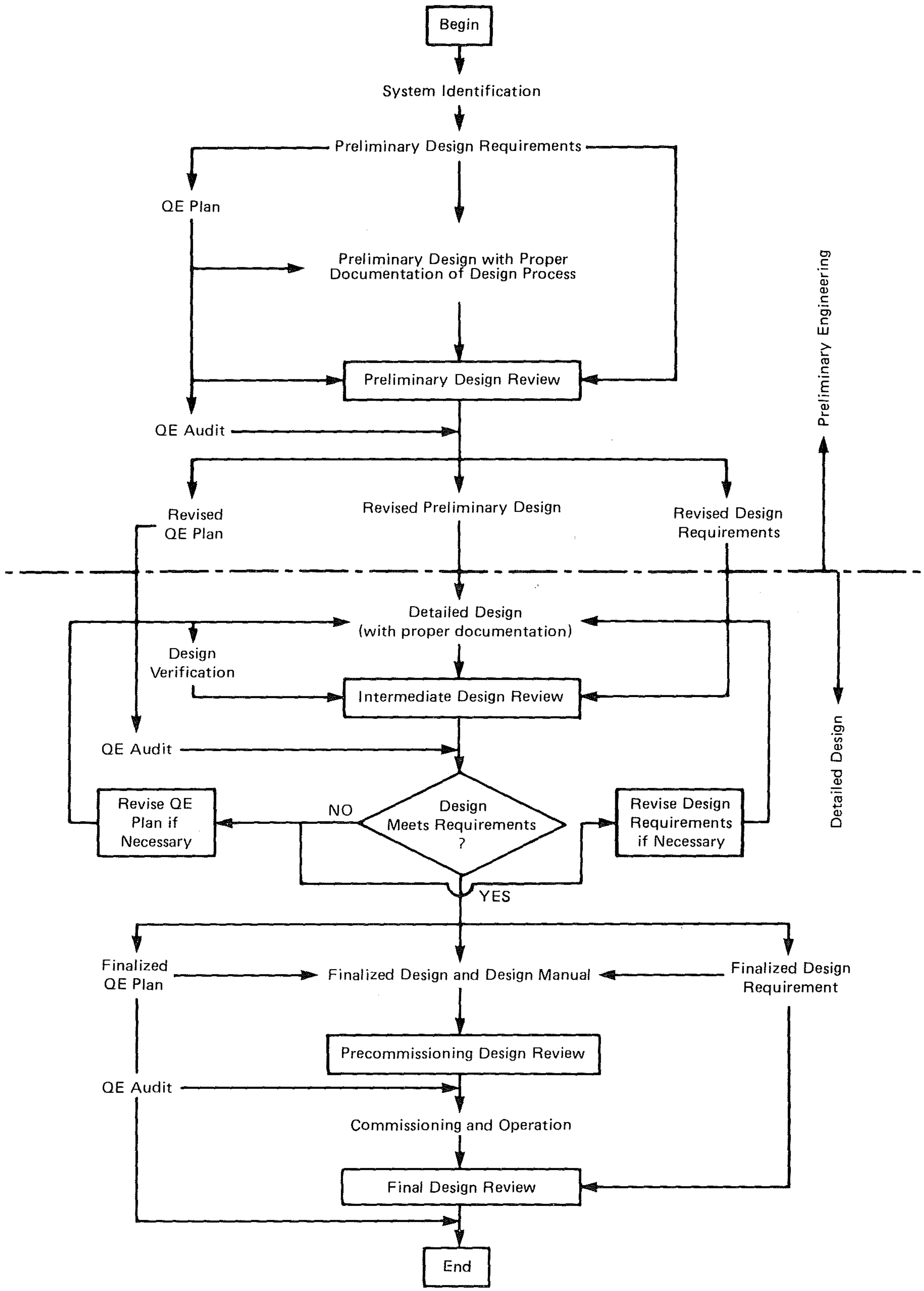


PROJECT LIFE CYCLE
CONCEPT
DEFINITION
Alternative schemes to meet power plan produced.
Board requirements determined
The specific plan approved.
PRELIMINARY ENGINEERING
– Requirements for major equipment and systems defined and specified.
– FUNDS RELEASED Major contracts awarded.
PROJECT PLANNING
– Work structured
– Supporting systems and structures defined
– Responsibility for work determined
– Schedules and estimates determined.
ACQUISITION
Design Procurement Constr. Comm.
Systems DESIGNED
Material PROCURED
Material CONSTRUCTED and installed
Material tested and functioning trails completed
OPERATION
Plant operated and maintained to produce commercial power until plant retired.
DECOMMISSIONING
GOAL
PRODUCE CAPACITY TO MEET LOAD GROWTH
PROJECT REQUIREMENTS
APPROVAL FOR PRELIMINARY ENGINEERING
PROJECT WORK REQUIREMENTS
APPROVAL TO PROCEED
MANAGERS APPOINTED
FIRST COMMERCIAL POWER

FIGURE 1-3

TABLE 1-1
Safety Design Guides
Darlington NGS

Title	Rev.
Purpose and Application of Nuclear Safety Design Guides	3
Common Mode Incidents – Overview and Design Requirements	3
Common Mode Incidents – Seismic Design	3
Limiting Consequential Damage of Postulated Pipe Ruptures	4
Shutdown Systems	2
Emergency Coolant Injection System	2
Containment	2
Extensions of the Containment Envelope	2
Environmental Qualification of Safety Related Equipment	1
Safety Assessments	0



*Note: Conceptual design stage, which precedes preliminary engineering, is not shown in this flow sheet.

FIGURE 1-4
Design Review Process

the decision making authority (usually Manager of Engineering) and the Design Review committee chairman is responsible for resolving any differences with the position taken by the decision making authority. A Follow-up Report is issued in a manner similar to the Design Review Report and any outstanding actions are then assigned to the department responsible for holding the design review. As illustrated in Figure 1-4, these design reviews are held at various stages throughout the design and construction phase to ensure that the actual design conforms to the requirements specified earlier.

The Nuclear Studies and Safety Department performs analyses and assessments to verify that the design meets its safety requirements and to demonstrate that regulatory requirements are met. The basis for the latter activity is the accident analysis documented in the Safety Reports for each nuclear station.

A sample of the accidents considered in the design process and reported in the Safety Report are shown in Figure 1-5. Note the combined process system failures with postulated failures in the Emergency Coolant Injection (ECI) system and in the containment envelope or containment subsystems. Consideration of such dual failures are unique to the Canadian licensing process.

All safety analyses and licensing submissions are reviewed in NSSD by senior staff members before external review. The review process is outlined in Figure 1-6 and involves senior staff in several departments and divisions including the Project Management, Directors of Design and Development - Generation, Nuclear Generation, Technical and Training Services, Health and Safety Divisions and various members of their staff. This review procedure ensures input from a wide range of perspectives and is implemented for all submissions to the AECB throughout the life of the project.

The involvement of the groups in the design area of Ontario Hydro's organization is not terminated when an adequately verified design has been demonstrated and a licence to operate a nuclear unit is obtained. There is an ongoing interaction between the design and operating groups to ensure that adequate nuclear safety is maintained throughout the life of a station. This aspect is discussed in the next section.

Similarly, groups in the operational area of the organization are involved throughout the design phase. Shortly after project commitment, a Station Manager is designated and he builds up his senior staff under attachment to the Project Manager. Later on, these staff form the cadre of the commissioning unit and finally of the operating staff. At about the same time, the AECB form a small review group headed by a senior project officer. This group, supported by a central staff in Ottawa, have regulatory responsibility for review of analyses, commissioning and operation for that station and moves permanently to the site prior to commissioning.

PROCESS SYSTEM FAILURES

- Fuel Handling
- Electrical Power Systems
- Loss of Regulation
- Reactor Components and Systems
- Heat Transport Loss of Coolant
- Steam and Feedwater System

PROCESS AND SAFETY SYSTEM FAILURES

Small and Large LOCA Combined with:

ECIS SYSTEM IMPAIRMENTS

- Loss of Injection
- Failure of Crash Cooldown
- Failure of Recovery Pumps
- ECIS Signal Failure
- Loss of Unit Electrical Power

CONTAINMENT SYSTEM IMPAIRMENTS

- Isolation Damper Failure
- Airlock Seal Failure
- Vault Cooler
- APRV Failure
- IPRV Failure
- Loss of Dousing

FIGURE 1-5
Accident Analyses Categories

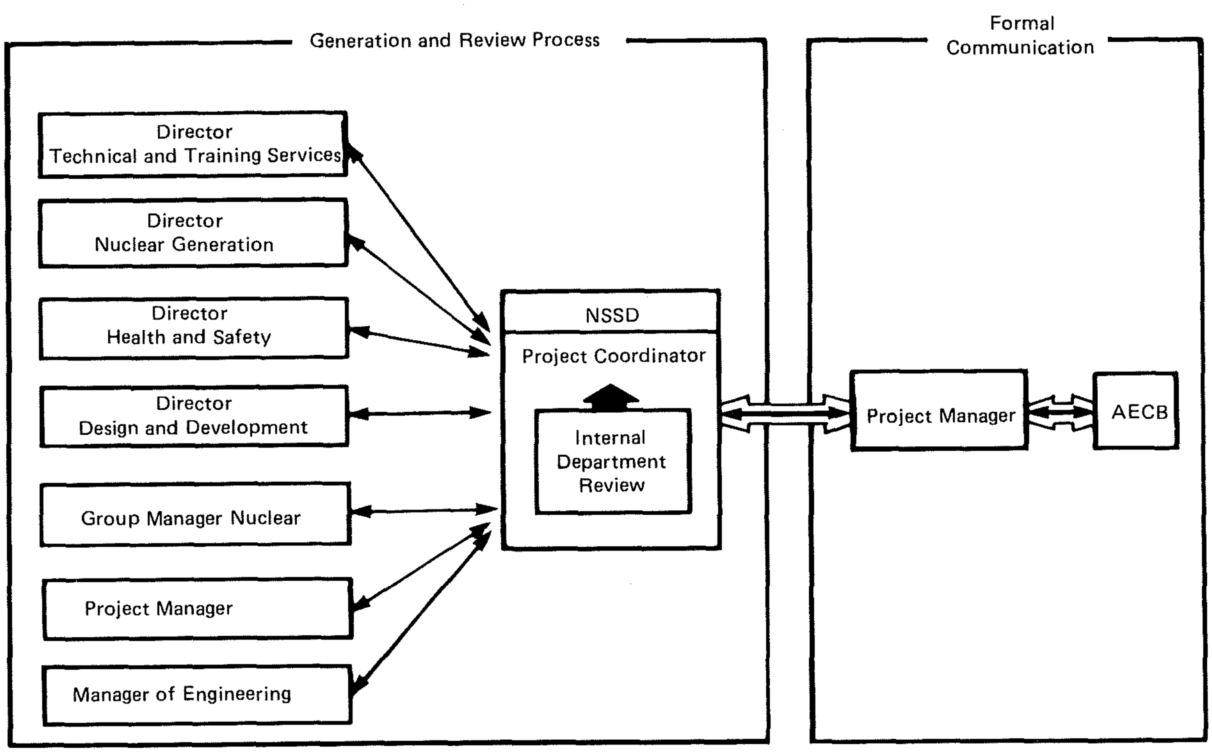


FIGURE 1-6
Safety and Licensing Documentation

1.3.2 Safety Management - Operational Phase

The Production Branch, shown in Figure 1-1, is responsible for commissioning and operating Ontario Hydro nuclear generating stations. This responsibility is delegated to the managers of the nuclear stations within the Nuclear Generation Division.

A typical station organizational structure is shown in Figure 1-7. The operating staff reporting to Production Manager are responsible for the actual operation of the plant. Day-to-day technical problems are the responsibility of the technical unit reporting to the Technical Manager. Technical problems requiring more extensive assessment are handled by support staff in the Technical and Training Services Division or by functional departments in Design and Development - Generation Division if they involve design modifications. The majority of maintenance operations at the station are handled by station maintenance staff - who report to the production manager. Training, business and planning staff report directly to the Station Manager.

Analogous to the generation of the Nuclear Safety Design Guides prepared by the design organization, the station manager issues a document called the Operating Policies and Principles (OP&P). This document specifies the principles and constraints to which the station will be operated and is subject to approval by the AECB.

One of the safety functions performed by the Technical Unit is the development of an "Abnormal Incidents Manual" (AIM). This manual is an operating document which is useful in helping the operator to minimize the risk to the general public and to station personnel following a safety system impairment, process system failure or common mode event. The document is developed after detailed review of the accident analyses and other safety studies and it provides an operator with guidelines and objectives for a variety of process system failure. Note that it provides procedures based on a wide sample of postulated accidents.

The document concentrates on ways of achieving the following broad objectives.

(a)	SHUTDOWN	Put the reactor in a safe shutdown state using the regulating system or the shutdown systems.
(b)	COOLDOWN	Ensure that an adequate heat sink is available for the removal of decay heat. Nuclear Generation Division (NGD) policy requires that for all operating modes there be a backup heat sink available.
(c)	DEPRESSURIZE	Reduce any leakage from heat transport systems.
(d)	CONTAIN	Ensure that no pathways are available for the release of radioactivity to the environment.

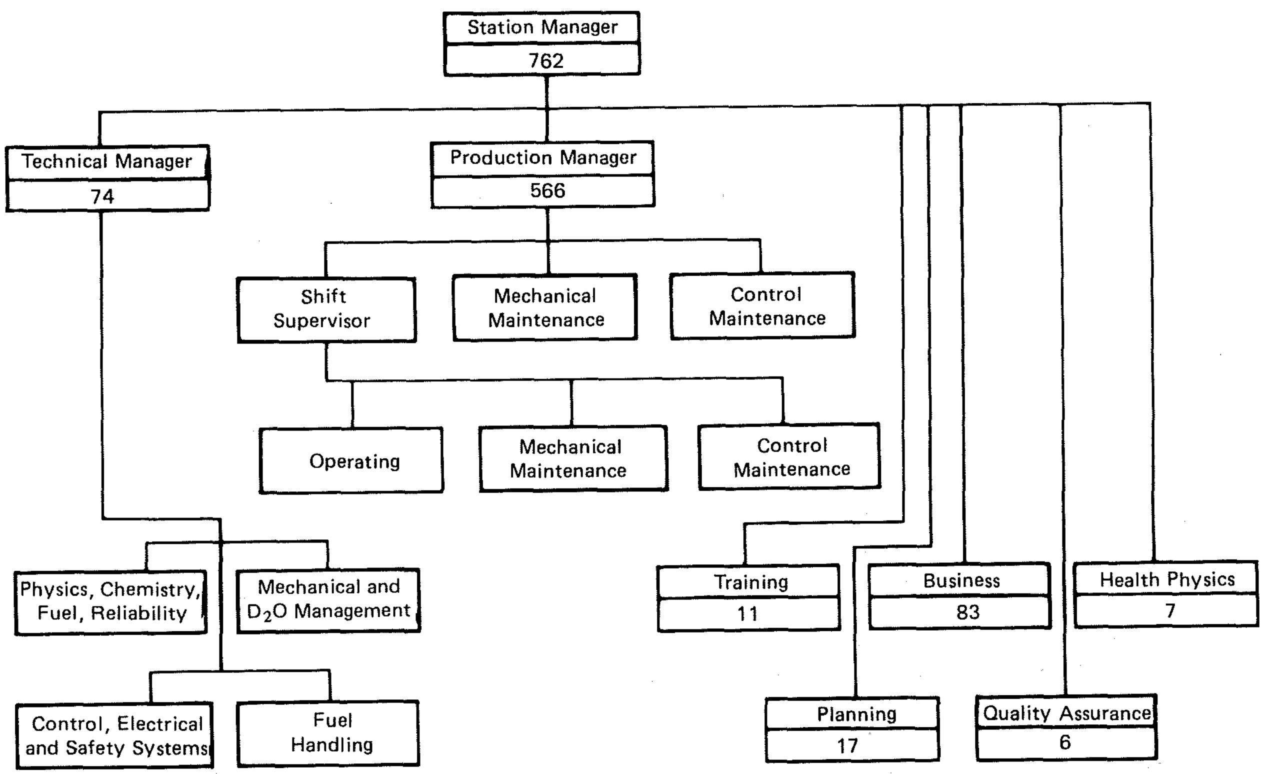


FIGURE 1-7
Bruce NGS A Organization

For a given abnormal incident, it authorizes the operator to take specified actions and/or initiate specified procedures. These procedures have been reviewed throughout the company and approved by the station manager and they thus relieve the operator of any worry he might have as to whether his superiors will approve of his actions. During an abnormal event the operator's full attention should be devoted to achieving the three objectives outlined above.

Prior to startup and during the early stages of low power operation, safety system commissioning tests are performed to demonstrate the functional characteristics of the systems and to demonstrate that assumptions used in accident analyses are conservative. AECB staff review of the commissioning tests results is a prerequisite to higher power operation.

A feature of the Canadian licensing philosophy is the requirement to demonstrate, on an ongoing basis, the high availability of the special safety systems. A detailed reliability assessment during the design of the plant establishes the testing frequency for the various system components. The resulting frequencies range from several tests per shift for shutoff rods, to tests every few years for passive components of the containment system.

When safety or process system faults occur, Significant Event Reports (SER) are generated immediately by the Operating Staff. The process for reviewing these SERs is illustrated in Figure 1-8. The process begins with a review by the shift supervisor, production manager and station manager of the event and the action of the shift operators. The technical unit is then responsible for follow-up action. This unit prepares a depersonalized version of the SER and arranges distribution of copies to the appropriate head office groups. The head office technical support groups provide assistance as required to the technical unit at the station and forward a copy of the SER to the Quality Engineering Department of the Design and Construction branch. They then forward these to the appropriate functional departments to determine if design action is required.

Routinely, the Nuclear Integrity Review Committee (see Section 1.1) meets to discuss safety matters related to operating plants. All safety related SER's are reviewed and policies and actions regarding safety system modifications and operating procedures are discussed, to ensure a unified approach to safety on all operating plants. The committee considers the adequacy of actions taken following the events, and initiates further action and deployment of additional resources where appropriate. The committee also monitors and updates the status of corrective actions resulting from previously reported events.

An important function for the ensurance of nuclear safety during the operational phase is the provision of well-trained personnel to staff the nuclear generating stations. The responsibility for this function rests with the Technical and Training Services Division in the Production

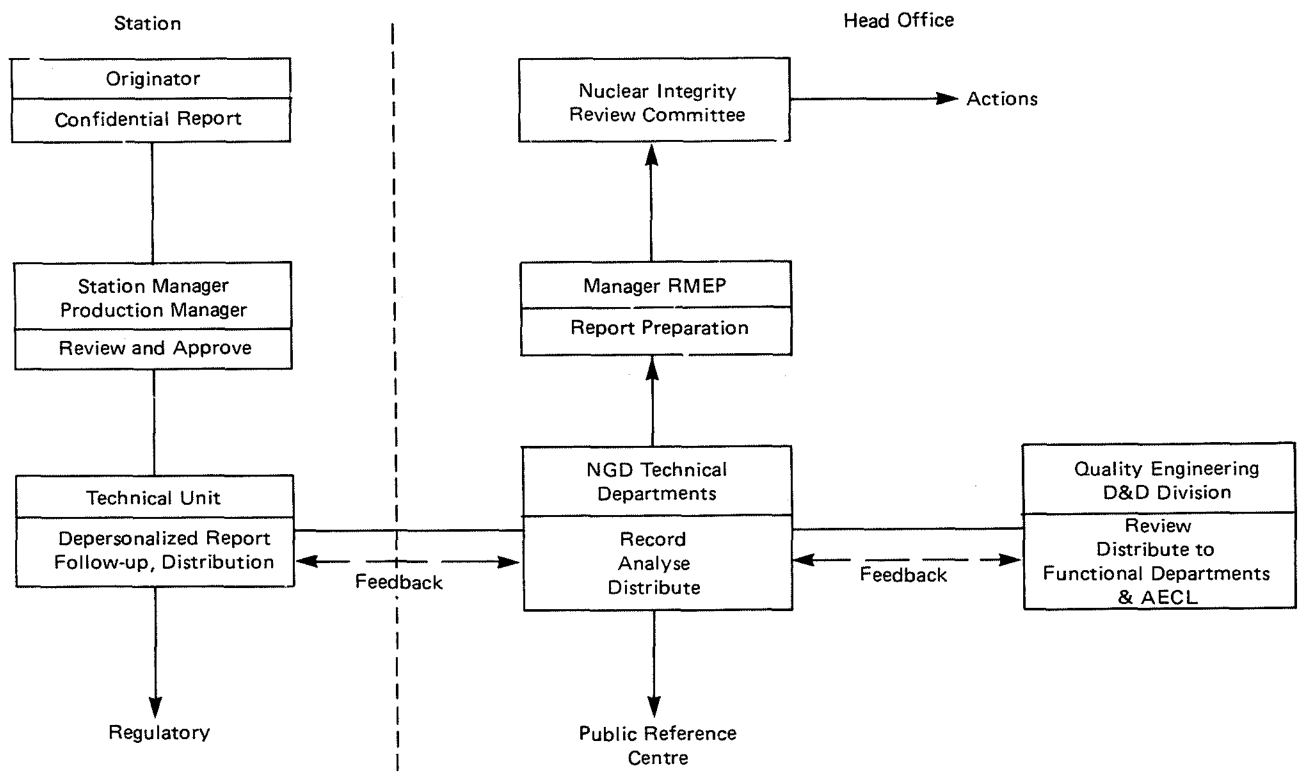


FIGURE 1-8
Significant Event Report Preparation and Distribution

Branch. Within this division there is a large training organization which trains a broad range of staff, from maintenance technicians to unit operators, supervisors and technical support staff. A full-scope training simulator is either operating or under construction for each multi-unit station. The subject of operator training is discussed in detail in Section 6 of this report.

1.4 Nuclear Emergency Preparedness

In the event of an accident at a nuclear station Ontario Hydro has the responsibility for management of all on-site remedial actions. The Ontario Provincial Government, through the Ministry of the Solicitor General, has the responsibility for all off-site actions deemed necessary to minimize any consequences to surrounding population.

There have always been plans in place both within Ontario Hydro and the Provincial Government for dealing with such contingencies. However, in 1983 the Government made major revision to the plan (Reference 6), with substantial input from Ontario Hydro and other affected organizations. More detailed procedures have been drawn up, involving local municipalities, other provincial government agencies and the Ontario Provincial Police, and various aspects of the plan are exercised annually.

This topic is dealt with in more detail in Section 13.

1.5 Licensing Process Overview and Responsibilities

1.5.1 General

The purpose of the licensing process is to show to the satisfaction of the AECB that the regulatory requirements will be met during the operation of the facility. The operating licence is a legal document which authorizes the utility to operate the plant in accordance with a set of conditions.

To achieve this it is necessary to identify all applicable regulatory requirements, incorporate those which are design-related into the design process and demonstrate by assessment and accident analysis that they are met, incorporate those which are operations-related into the commissioning/operating program and by means of documentation show that this has been done. During the operating phase, part of this process continues to ensure that the conditions of the operating licence are maintained.

1.5.2 AECB Licensing Requirements

The Canadian regulatory framework is founded upon the principle that nuclear energy production should produce risks that are lower than those of equivalent existing industries. AECB licensing requirements have evolved over the past three decades and have been published in various AECB papers and discussed in great detail between the utilities, AECL and AECB.

In general, however, most of Ontario Hydro's nuclear plants have been licensed in accordance with the 1972 version of the Siting Guide (Reference 7). The acceptance criteria of the Siting Guide are illustrated in Table 1-2.

The Canadian defense-in-depth approach inherent in the Siting Guide leads to the following requirements:

- The design and manufacture of components and the design and construction of systems and structures must be in accordance with nationally or internationally accepted engineering codes and standards;

- Process and special safety systems must be physically and functionally separate, diverse and independent to the extent practicable;

- Process and special safety systems must be testable and testing programs must be implemented such as to demonstrate continued capability to meet performance specifications;

- Process and special safety systems must be designed to be fail-safe to the extent practicable;

- Redundancy is required in all safety-related control, monitoring and initiation systems; and

- The selection and training of operations personnel must be such as to ensure continued compliance with the high standards which have characterized nuclear operations in Canada.

Other requirements have been defined in the course of licensing the Pickering and Bruce generating stations. These have been assembled in "consultative documents" (Reference 8) C-7, C-8 and C-9 pertaining to requirements for containment, shutdown and emergency core cooling systems respectively. In addition, the various requirements which have been imposed on the licencees throughout the licensing history in Canada have been compiled in draft consultative document C-6 (Reference 9), pertaining to the manner in which accident analyses are to be performed.

The criteria documented in AECB Consultative Document C-6 (Reference 9) are illustrated in Table 1-3.

TABLE 1-2
AECB Siting Guide (1972)
Operating Dose Limits and Reference Dose Limits for Accident Conditions

Situation	Assumed Maximum Frequency	Meteorology to be Used in Calculation	Maximum Individual Dose Limits	Maximum Total Population Dose Limits
Normal Operation		Weighted According to Effect, i.e., Frequency Times Dose for Unit Release	0.5 Rem▼/Yr Whole Body 3 Rem/Yr to Thyroid	10^4 Man-Rem/Yr 10^4 Thyroid Rem/Yr
Serious Process Equipment Failure*	1 per 3 Years	Either Worst Weather Existing at Most 10% of Time or Pasquill F Condition if Local Data Incomplete		
Process Equipment Failure Plus Failure of Any Safety System	1 per 3 x 10^3 Years	Either Worst Weather Existing at Most 10% of Time or Pasquill F Condition if Local Data Incomplete	25 Rem Whole Body 250 Rem Thyroid	10^6 Man-Rem 10^6 Thyroid Rem

* A serious process failure is defined as any failure of process equipment or procedure which requires action of a special safety system to prevent significant fuel failures or significant radioactive material releases from the station.

▼100 Rem = 1 sv

TABLE 1-3

Radiation Dose Limits to a Member of the Public
ofr an Event of a Given Frequency*

Event Class	Qualitative Event Frequency Criteria	Expected Event Frequency (occurrence/reactor-yr)	Individual Dose Limit Whole Body (rem)†	Individual Dose Limit Thyroid (rem)
1	Greater than 50% chance of occurring in the lifetime of a single reactor, or More frequency than twice in the lifetime of a four-unit station	$f > 10^{-2}$	0.05	0.5
2	About once in the lifetime of an eight-unit station	$10^{-2} > f > 10^{-3}$	0.5	5
3	Low probability postulated failure	$10^{-3} > f > 10^{-4}$	3	30
4	Very low probability postulated failure	$10^{-4} > f > 10^{-5}$	10	100
5	Extremely low postulated failure	$f < 10^{-5}$	25	250

* This table is based on AECB Consultative Document C-6 "Requirements for the Safety Analysis of CANDU Nuclear Power Plants". The licensing criteria for Darlington NGS are based on this table.

† 1 Sv = 100 rem

The Darlington Generating Station is being licensed on the basis of trial application of the above consultative documents and not the Siting Guide as were other plants in Canada. To replace the reference dose limits of the Siting Guide, for Darlington a five event class arrangement has been agreed to between Ontario Hydro and the AECB with a dose limit for each event class. Ontario Hydro have issued the event frequency ranges shown in Table 1-3 as an aid to event classification.

1.6.3 Construction Approval and Operating Licence

To obtain construction approval it is necessary to submit documentation (in particular the accident analysis) to show that the chosen concept, with substantial detailed design still underway, will be able to meet the regulatory requirements. The main vehicle for this is the preliminary Safety Report. Although the regulatory requirements have by then been fed into the design process, clarification is often sought by the AECB staff on how the requirements are being implemented in detail. This leads to continuing dialogue throughout the design phase. The AECB grants an operating licence when it is satisfied that regulatory requirements are adequately met. Any residual issues dealt with through the application of conditions in the operating licence, and the resolution thereof is monitored. The licensing process is described in more detail in Reference (10).

To facilitate the process of meeting the commitments made during the licensing process, Ontario Hydro has established and maintains a register of all important licensing documentation for all its nuclear facilities.

Ontario Hydro's overall performance on meeting regulatory requirements is continually under review by AECB staff resident at the station and supported by their head office staff. There is an ongoing interaction between this staff and those Ontario Hydro staff involved in the safety management program. Formal annual reviews of the station performance are conducted by the AECB, and Operating Licences, subject to periodic renewal, are reissued upon demonstration of satisfactory performance.

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Safety Philosophy and Design Approach

2.0 SAFETY PHILOSOPHY AND DESIGN APPROACH

2.1 General

The foundation of our safety philosophy, sometimes termed "defence-in-depth", is established by designing, constructing and operating a nuclear power plant in such a manner that the radioactive materials are contained within a succession of physical barriers (References 1,2,3,4,5). This basic concept, similar to that adopted in other countries (References 7-12), recognizes that in spite of the best efforts of designers and operators, equipment failures and operator errors may occur such that one or more barriers to radioactivity release is breached. The safety approach, therefore, requires that more than one "level" of safety be provided for each barrier for a range of postulated accidents that may threaten the barrier. This leads to a multi-barrier, multi-level "defense-in-depth" system which is uniquely tolerant to a wide range of equipment failures and human error. This overall safety design approach, including the special safety features, has led to an excellent safety record in operating nuclear generating stations and provides the confidence that risk from these stations can be maintained at a low level.

The safety approach presented here is intended to provide a concise account of the basic principles which govern the design requirements of a nuclear power plant. The safety approach (Reference 13) and its application to requirements of design for safe operation, safety analysis and accident management constitute a code of practice that is internationally accepted. The major safety principles thus derived are as follows (Reference 13):

(a) The principal goal of nuclear safety shall be to keep radiation exposure of the public and the site personnel within appropriate prescribed limits* in all operational states and within acceptable limits* in accident conditions;

(b) Nuclear power plants shall be designed, constructed and operated to an appropriate quality level to minimize the likelihood of a failure which could lead to the escape of significant quantities of radioactive material;

(c) Escape of radioactive material shall be restricted by successive physical barriers, some of which are present because they have functions to perform in power operations, others serving safety purposes only;

(d) Consideration shall be given to all postulated initiating events* that may render any of these barriers ineffective;

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* For definitions and details, see Appendix and Annex of Reference 13.

(e) All design provisions and necessary operator actions which are required to:

(1) maintain intact those barriers which prevent the escape of radioactive material, or

(2) mitigate the consequences of barrier failure.

shall be identified for each postulated initiating event and shall be part of the design basis for the plant.

The purpose of the safety approach, as highlighted here, is to maintain the plant in a normal operating state. In addition, the objective is to ensure the correct short term response following an unanticipated event and to facilitate the long term management of the plant following accident conditions. To be quite specific, the criteria for nuclear power plants in Canada which flow from this general approach include the following (Reference 14):

(1) the design and manufacture of components and the design and construction of systems and structures must be in accordance with nationally or internationally accepted engineering codes and standards;

(2) process and special safety systems must be physically and functionally separate, diverse and independent to the extent practicable;

(3) process and special safety systems must be testable and testing programs must be implemented such as to demonstrate continued capability to meet performance specifications;

(4) redundancy is required in all safety-related control, monitoring and initiation systems; and

(5) the selection and training of operations personnel must be such as to ensure continued compliance with the high standards which have characterized nuclear operations in Canada.

The multiple barrier principle (Reference 2 and 3) is an important element in the philosophy of safe nuclear plant design. In its simplest form, the barriers are as follows:

(a) the uranium dioxide (UO_2) fuel, which contains almost all the radioactivity, is a ceramic with high melting point sealed in a corrosion resistant metallic cladding;

(b) the zirconium alloy fuel element sheath;

(c) the reactor cooling system;

(d) the containment system capable of containing most of the radionuclides if the above fail;

(e) exclusion boundary which provides a separation between the station and the public.

One representation of the levels of safety provided in the CANDU nuclear operating station is illustrated in a matrix form in Figure 1. The first two levels of safety, in general, are capable of dealing with a large majority of safety concerns. However, the third level receives the most attention since the postulated failures at this level are not only more serious, although low frequency events, but they are also used to set the design bases for the special safety systems. Designing the special safety systems to deal with these major, extreme accidents leads to large margins of safety for more likely events.

The aim of the first level of defence is to prevent deviation from normal operating conditions. This requires that the plant be soundly and conservatively designed, constructed and operated in accordance with appropriate quality levels and engineering practices. To meet this objective, significant efforts are directed towards selection of appropriate design codes and materials, and to the control of component fabrication and of plant construction. In addition, the principles of redundancy and diversity are applied to enhance reliability of the design. Redundancy requires the use of more than the minimum number of sets of equipment to accomplish a given safety function, such that failure or unavailability of one set of equipment can be tolerated without loss of safety function. The principle of diversity is applied to reduce the potential for common cause failures by incorporating different attributes into the systems or components.

The aim of the second level of defence is to detect, in a timely manner, and intercept deviations from normal operating conditions in order to prevent an operational occurrence from escalating into an accident condition. This recognizes the fact that during the service life of a nuclear power plant, events will occur which will challenge the integrity of the system. This level, therefore, requires the provision of specific systems and definition of operating procedures to prevent or minimize damage from anticipated initiating events.

For the third level of safety it is assumed that, although very unlikely, the escalation of certain events may not be arrested by a preceding level and so additional safety systems, equipment and procedures are provided to control the consequences of the resulting accident conditions.

It is these levels of safety, when integrated into quality assurance programmes for all phases of the establishment of a plant, provide the necessary confidence that the overall safety approach is sound. The capability built into Ontario's CANDU generating stations, to meet the intent of the "defense-in-depth" concept is elaborated further in the sections below.

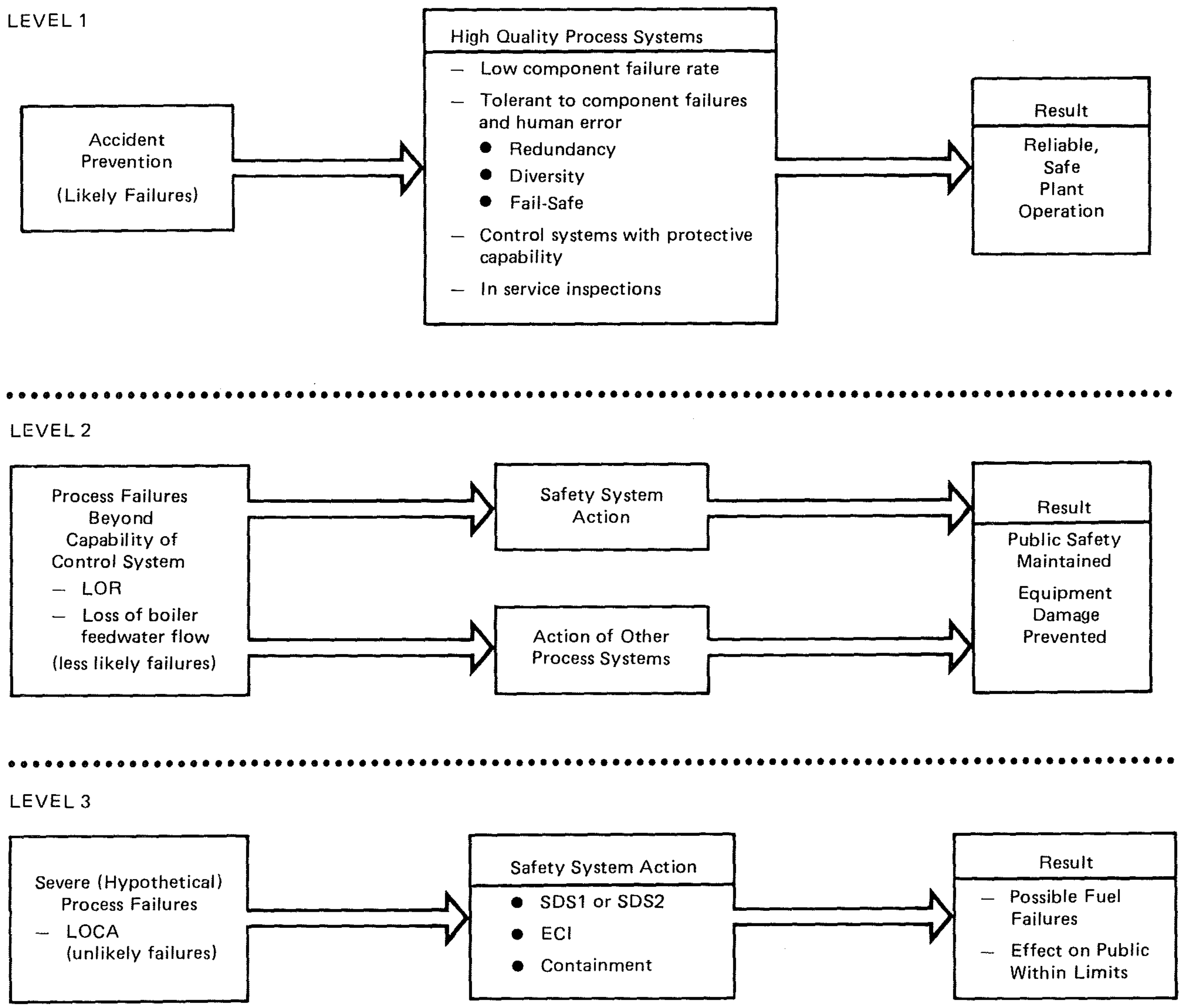

FIGURE 2-1
Levels of Safety

2.1.1 Equipment Quality

The first and foremost level of safety provided in nuclear generating stations is the design, construction and operation of the nuclear process systems to a high quality level. Process systems, such as the heat transport system, are designed to codes and standards which demand the highest quality in design, material and workmanship. These standards were developed for the nuclear industry and they reflect a much higher level of quality than is applied to conventional process systems. In addition, these systems undergo an extensive in-service inspection and maintenance program to provide continued assurance of a high level of integrity. The Quality Engineering (QE) and Quality Assurance (QA) programme documentation provided the evidence that the design, supply, construction and operating practices conform to the required standards.

2.1.2 Process and Control Systems

The plant process and control systems are designed to be able to cope with a range of anticipated failures in plant equipment. This is achieved through automatic control action or starting standby equipment, or by regulating systems that can automatically adjust setpoints to accommodate changed system conditions. Such actions prevent the failures from degenerating into serious accidents. These systems are not unlike those provided in some process plants where high reliability is an economic necessity. Active systems such as the reactor regulating system are designed with redundant components so that many single equipment failures, and in some cases multiple equipment failures, will not cause a loss of the process function. In addition, for many of those failures which can lead to a loss of the process function, the result will be a safe action. The dual computer control system is an example of this approach. Each computer processes the input information and provides an output of a control action based on these inputs. If the computer fails to execute the necessary steps, this failure is detected and control is transferred to the other computer. If this computer similarly fails, all control output signals are set to a safe value, resulting in a reactor shutdown.

2.1.3 Special Safety Systems

In addition to the normal process systems that contribute to the power production process are a number of special safety systems that will ensure the capability to meet safety targets. These are described in Section 3.

2.1.4 System Independence

In the above approach, it is important that adequate independence between the process systems and the safety systems and between safety systems be demonstrated. There must be no causal propagation of the initial failure into other levels of defense, that would jeopardize the defense-in-depth capability. Furthermore, there must be no credible single event that can cause a serious process failure and at the same time a failure of the required safety provisions to respond to the process failure (common cause failure).

2.1.5 Human-Machine Interface

Systematic consideration of human factors and the human-machine interface is an integral part of the design process which begins at the conceptual design stage and continues throughout the operational phase. The objective is to provide the operator clear displays of parameters that indicate the status of all equipment and systems necessary to achieve safe operation, or shutdown, if required, and to provide the controls to allow the required operator response. The design principles for information display and controls aim to promote the success of operator actions in the light of the time available, the expected physical environment and the psychological pressures. The need for operator intervention on a short-time scale is kept to a minimum, and the design generally takes into account the fact that such intervention is only acceptable if it can be shown that the operator has sufficient time to decide and to act.

2.1.6 Design Evaluation

The final level of assurance is to carry out a detailed review of a wide range of potential failures in the plant to determine the effectiveness of the protective features. This enables an estimate of public risk to be made and allows designers to direct their attention to those areas which could potentially affect this risk level. Details are discussed in Section 4.

2.2 Development of System-Level Safety Design Requirements

2.2.1 General Safety Design Intent

The general safety design intent for nuclear stations is to provide a design which will

(a) ensure that radiation doses to the public and site personnel do not exceed prescribed limits for any operational state;

(b) meet AECB licensing criteria.

The process of determining the nuclear safety requirements of the various safety related systems is an iterative process that involves:

(a) identification of events of concern;

(b) identification of provisions which will be required to cope with each event, along with their functional and performance requirements with respect to their safety-related role;

(c) verification that the design of safety related systems can meet their safety related requirements, and that their integrated effect will lead to the meeting of the licensing requirements and the intent of the safety approach.

Functions (a) and (b) will be elaborated below. Function (c) is described in Section 4.

2.2.2 Event Identification and Classification

The objective of this activity is to identify all events that will be considered in the design as being capable of leading to radiation exposure to the public.

On Pickering and Bruce the approach has been to identify (as per the Siting Guide) all serious process equipment failures (single failures) and all serious process equipment failures plus failure of any Special Safety System (dual failures) (see Table 1-2 for definitions).

On Darlington a more systematic approach was taken to identify all events which can lead to public exposure that fall within the frequency range of 10^{-5} per year and higher (Table 1-3). Risk significant events lower than 10^{-5}/yr and higher than 10^{-7}/yr are also considered on a case by case basis. Events below a frequency of 10^{-7}/yr are considered incredible.

In order to achieve this, both initiating events and combinations of these with failure of mitigating provisions must be identified.

There are two general categories of initiating events: internal events and external events. Internal events are abnormal conditions generated within the plant as a result of equipment failure or human error. External events are natural and man-made phenomena originating outside the plant which have the potential for causing wide-spread, multiple internal events.

To identify all possible types of internal initiating events a system-by-system review is carried out to identify those containing significant quantities of radioactive materials. For each such source of radioactive materials, it is possible,based on the knowledge of the plant processes and past experience in selecting initiating events, to determine all possible ways in which unplanned release of these materials can occur. This process provides a comprehensive list of internal initiating events. (Refer to Section 4, Figure 4.1).

External initiating events are identified through a comprehensive survey of natural phenomena and industrial activities in the vicinity of a plant. The severity selected to characterize these events was generally derived to cover the frequency range of 10^{-5}/yr and higher. Typical external events considered in the design of Darlington NGS are listed in Table 2-1.

To complete the list of abnormal events, all combinations of initiating events and compounding failure of relevant mitigating provision(s), which fall within the frequency range of 10^{-5}/yr and higher are identified.

In addition to the process systems, which are the systems required for electric power production, plant design includes a number of provisions made for the sole purpose of mitigating the effects of abnormal events.

The equipment associated with these provisions is normally in standby state and can respond to an abnormal event by manual or automatic initiation. Standards to which these provisions have been designed and constructed and to which they will be tested and maintained, vary according to their importance in limiting the public health and economic risk. Special safety systems are mitigating provisions of highest standard.

Provisions fall within one of the following categories:

(a) the capability to shut down the reactor;

(b) the capability to limit overpressure in the Heat Transport System;

(c) the capability to remove decay heat;

(d) the capability to limit release of radioactive materials.

More specific identification of mitigating provisions, and their specific role in providing one of the above capabilities, is given in Section 3.

Based on the knowledge of plant behaviour following an initiating event, it is possible to identify event combinations involving initiating events and failure on demand of one or more of the relevant mitigating provisions.

The degree of any physical or functional cross-linking between the initial failure and relevant mitigating provisions themselves is considered in deriving the frequency of combinations and is reflected in the classification of the combination. In many instances, however, the beneficial effect on the frequency and the consequence of an accident of some of the mitigating provision is discounted in order to limit the scope of the accident analysis to a manageable level. Combinations involving simultaneous failure of two or more independent process systems do not usually meet the event selection criteria because of the very low probability of such occurrence. An independent occurrence of a second process system failure in the long term following an initial occurrence is of higher likelihood. However, because of the plant being in a shutdown state, the effect of this second failure is minor or inconsequential.

Additional process failures following the initiating events are included in the analysis whenever such failures are a direct result of the initiating event either through a functional or physical crosslink.

The simultaneous occurrence of internal and external events is not considered because of its low probability. Where internal events can realistically be caused by external events these are considered along with the safety assessment of the external event.

Table 2-1

TYPICAL COMMON-MODE EVENTS CONSIDERED IN STATION DESIGN

On-site explosion

Off-site explosion

Turbine breakup

Internal fires

Design basis tornado

Design basis earthquake

Airplane crash

Flooding

Table 2-2

Mitigating System Failures Considered

System	Function
Shutdown Cooling	decay heat removal
Auxiliary Feedwater	feedwater supply to the steam generators
D_2O Recovery/D_2O Feed	coolant make-up to HT system
Post Accident Cooling Water	cooling of HT coolant
Steam Generator Emergency Cooling	short-term feedwater supply to steam generators
Emergency Service Water	long-term feedwater supply to steam generators
Moderator System	ultimate heatsinks for core decay heat
Condenser Steam Dump Valves and Atmospheric Steam Dump Valves	steam release from steam generators

Combinations of independent low probability external events are not considered because of their extremely low probability of occurrence. External events which could realistically be caused as a result of another external event are considered along with the initial event.

Mitigating system failures that are considered in combination with initiating events are listed in Table 2-2. Failures or impairment of special safety systems are also considered. Examples of these impairments are shown in Table 2-3.

2. Safety Design Requirements for the Safety Related Systems

For each event identified above, a safety design objective is established which, if met, will allow the overall safety objective to be met (i.e., no fuel failures for class 1 and 2 events, fuel channel integrity large LOCA, etc). From these, and from the general knowledge of plant response following abnormal events, functional and performance requirements, including the reliability requirements, are developed for each system involved. The totality of these safety requirements are recorded in documents called "Nuclear Safety Design Guides". These requirements are eventually recorded for each system, along with other requirements in the system's "Design Requirements" document. A list of typical Design Guides is shown in Section 1, Figure 1-3.

Table 2-3

Special Safety System Failures Considered

System	Function
Emergency coolant injection	coolant make-up to HT system
Instrumented steam generator safety relief valves	steam release from steam generators
Containment Envelope	isolate containment
Airlock and transfer chamber seals	isolate containment
Containment isolation dampers	isolate containment
Main containment pressure relief valves	connect containment to vacuum building
Instrumented relief valves	connect containment to vacuum building
Auxiliary relief valves	connect containment to vacuum building
Reverse Flow valves	connect containment to vacuum building
Reactor vault coolers	cool containment
Filtered air discharge	remove activity from exhaust air

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Safety Design Features

3.0 SAFETY DESIGN FEATURES

3.1 Introduction

The design of nuclear power plants incorporate a multitude of features which contribute to their safe and reliable operation. The design basis is that while we design systems to very high standards failures in equipment or procedures will eventually occur and provisions are, therefore, incorporated in the design to render these failures inconsequential or to minimize their effects on the plant and the public beyond the boundary of the plant. In addition to fault tolerant design features built into the basic processes and systems that are used in power production, there are a number of systems, called Special Safety Systems, that can act to mitigate the consequences of failures that are beyond the capabilities of the normal plant equipment or in the eventuality that normal plant equipment does not function properly. The plant design must be shown to be capable of limiting offsite effects even when each of these special safety systems are postulated to be impaired to some extent.

The design of the plant thus incorporates a defense in depth approach which comprises a high quality design that minimizes failures, design features to mitigate the effects of failures and special safety systems that can accommodate failures in other lines of defense.

The designs of the various Ontario Hydro nuclear generating stations has evolved over the years. Although each reactor has a basic CANDU reactor core, the auxillary equipment and its arrangement has changed from station to station. A table summarizing some main design features of each station is shown in Table 3-1.

3.2 Process Systems

Process systems is a term used to describe those systems that are used in the normal operation of a nuclear power plant. The process systems that are particularly important to nuclear reactor safety are those that are associated with transporting the heat away from the reactor fuel to the point where it can be harnessed to generate electricity or can be rejected to the surrounding environment, and those associated with controlling the power level within the fuel. Maintaining an adequate balance between the rate at which heat is generated within the fuel by the fission process and the decay of fission products and the rate at which heat can be transported from the fuel will maintain two initial barriers to the release of fission products to the environment. These initial two barriers are the ceramic-like uranium dioxide fuel matrix in which most of the fission products are trapped and the zirconium alloy fuel sheaths which protect the fuel itself from erosion by the coolant and retain fission product gases which escape from the fuel matrix under normal operating conditions.

TABLE 3-1
Comparison of Pickering GS A, Bruce GS A, Pickering GS B, Bruce GS B and Darlington GS A

	Pickering GS A	Bruce GS A	Pickering GS B	Bruce GS B	Darlington GS A
General					
Unit electrical output Mwe (net)	515	740	516	837	881
Number of units	4	4	4	4	4
First unit in service date	1971	1977	1983	1984	1988 (Sched)
Core and Fuel Data					
No. of fuel channels	390	480	380	480	480
No. of fuel bundles/channel	12	13 (12 in core)	12	13 (12 in core)	13 (12 in core)
No. of elements/bundle	28	37	28	37	37
Nominal design channel power (MWth) (without ripple)	5.1	6.5	5.37	6.5	6.4
Nominal design bundle power (kWth)	640	900	650		
Nominal outer element					
• linear power (kW/m)	52.4	57.3	53.4		
• $\int \lambda d\theta$ (kW/m)	4.2	4.2	4.25	4.2	4.2
Reactor					
Calandria vault	Air filled	Water filled steel shield tank	Water filled steel-lined concrete	Natural water filled steel shield tank	natural water filled steel shield tank
End shielding	Steel plates and water	Steel balls and water	Steel balls and water	Steel balls and natural water	Steel balls and natural water
Min. pressure tube wall thickness, (mm)	4.06	4.06	4.06	4.2	4.2
Reactivity Mechanisms					
Adjuster rods (absorbers)	8(U1&2)/18(U3&4) (Cobalt)	–	21 (Cobalt)	24 (Cobalt)	24 (Stainless Steel)
Booster rods (enriched uranium)	–	16.0	–	–	–
Control absorbers	–	4 (Cadmium)	4 (Cadmium)	4 (Cadmium)	4 (Cadmium)
Vertical flux detectors	8	20	34	27	23
Horizontal flux detectors	–	12	25	8	14
Safety Systems					
Primary shutdown	21(U1&2)/11(U3&4) Shut-off rods augmented by moderator dump	30 Shut-off rods	28 Shut-off rods	32 Shut-off rods	32 Shut-off rods
Shutdown rate (Time to full insertion) (sec)	1.8	1.8	1.8	1.8	1.8
Secondary shutdown	–	7 Poison injection nozzles	6 Poison injection nozzles	8 Poison injection nozzles	8 Poison injection nozzles
Emergency coolant injection (ECI)	U1&2 from high pressure pump system (SES power), then from moderator pumps. U3& U4 from moderator system	Fast acting – high pressure (5.8 MPa) accumulators Long term – pumps	From high pressure pump system (SES power)	Fast acting – high pressure (5.8 MPa) accumulators Long term – pumps	From high pressure pump system (Class IV power)
Reactor building design pressure kPa gauge (psig)	41 (6)	69 (10)	41 (6)	82.7 (12)	96.5 (14)

TABLE 3-1 (Cont'd)
Comparison of Pickering GS A, Bruce GS A, Pickering GS B, Bruce GS B and Darlington GS A

	Pickering GS A	Bruce GS A	Pickering GS B	Bruce GS B	Darlington GS A
Heat Transport System					
Inlet temperature °C (°F)	249 (480)	252/265 (483/509)	249.5 (480)	250/265 (482/509)	266 (511)
Outlet temperature °C (°F)	293 (560)	304 (579)	203.4 (560)	305 (581)	310 (590)
Quality at outlet header	5°C subcooled	Subcooled†	7°C subcooled	Subcooled†	2.0%
Inlet pressure, core MPa absolute (psia)	9.54 (1 384)	10.32 (1 497)	9.55 (1 386)	10.6 (1 542)	11.4 (1 652)
Heavy water inventory m^3/MWe (ft^3/MWe)	0.317 (11.2)	0.385 (13.6)	0.317 (11.2)	0.283 (10.1)	0.247 (8.73)
No. of steam generators	12	8	12	8	4
Surface area per steam generator m^2 (ft^2)	1 858 (20 000)	2 369 (25 000)*	1 843** (20 000)	2 400 (26 000)	4 830 (52 415)
No. of pumps	16 (4 standby)	4	16 (4 standby)	4	4
Pump motor power:					
Operating power MW(e) (HP)	1.23 (1 650)	6.41 (8 600)	1.17 (1 575)	6.41 (8 600)	6.70 (8 989)
Rated power MWe(e) (HP)	1.42 (1 900)	8.20 (11 000)	1.42 (1 900)	8.20 (11 000)	9.40 (12 610)
Pressure control	feed and bleed	pressurizer	feed and bleed	pressurizer	pressurizer
Purification half life, min.	130	65	130	60	60
Moderator System					
No. of pumps	5 at 25%	2 at 100%	5 at 25%	2 at 100%	2 at 100%
Heavy water inventory m^3/MWe (ft^3/MWe)	0.535 (18.9)	0.425 (15.0)	0.479 (16.9)	0.396 (14.1)	0.400 (14.1)
Secondary System					
Steam flow Mg/s (10^3 lb/h)	0.81 (6.46)	1.17 (9.3)+	0.81 (6.47)	1.17 (9.3)+	1.32 (10.5)
Steam pressure (drum) MPa absolute (psia)	4.09 (593)	4.38 (635)	4.09 (593)	4.38 (635)	5.07 (735)
Net station efficiency (%)***	29.1	29.1	29.1	32.0	30.4

* Includes separate preheater
** Based on 2573 tubes
*** Net electrical output/6otal fission power

\+ Excluding flow to steam transformer plant.
† Some channels have up to 4% quality.

To ensure that the coolant remains available to the fuel, the heat transport system is designed and constructed to the highest standards for nuclear grade pressure-retaining systems and components (CSA Standard CAN3-N285.1 Class 1). This standard, which incorporates all the requirements of the ASME Code Section III, ensures that the system will be designed with generous margins in allowable stresses, that the highest quality materials that are traceable to their source are used, that the system will be assembled by skilled craftsmen that must undergo stringent certification requirements, that the system will be 100% inspected during construction to ensure initial integrity, and that a portion of the system will undergo a periodic inspection while in service to ensure that there is no systematic degradation of the system during its operating life.

The primary system piping is a very tough material, ASME SA106 Grade B seamless carbon steel. Analysis and material testing has shown that a crack or flaw in the piping of minimum detectable size would not grow through stress cycles to through wall size during the life of the plant. Even if a crack were postulated to penetrate the piping, fracture mechanics analysis has shown that the crack would leak at a rate that is many times the detectable level before it would reach a length where it could propagate in an unstable manner (critical crack length). This is the so called "leak-before-break" phenomenon.

There is a continual make-up to the heat transport system by redundant pressurizing pumps or feed pumps to account for leakage from pump seals, valve packing, etc, as well as bleed flow to the purification system. These pumps can draw from a heavy water storage tank to provide make-up for sizeable leaks that could occur during operation. Recovery pumps are provided to return collected coolant to the heavy water storage tank. It was these process systems that were used to cope with the pressure tube failures at Pickering Unit 2 in 1983 and at Bruce Unit 2 in 1986.

Heat can be removed from the core to the steam generators while at power and to either the steam generators or a shutdown cooling system while at low power. Under shutdown conditions, heat can be transported to the steam generators by thermosyphoning if the heat transport pumps are not operating, a phenomenon demonstrated by commissioning tests on each reactor design.

3.3 Plant Control

3.3.1 Regulating Systems

A feature of the design of CANDU power stations is their powerful, sophisticated, redundant, self checking integrated control systems. CANDU generating stations were among the first in the world to take advantage of the capabilities of the speed and versatility of digital computer control. The configuration generally used is to have one computer in control and the second computer ready to assume control should the first fail. Many of the signals fed to the computers, and especially the important signals are duplicated or triplicated so that the signals can be compared to ensure consistency or to employ some form of majority vote logic. In general, the systems are designed to fail to what is considered a safe state in most circumstances or to transfer control when signals are ambiguous.

The main role of the reactor regulating system is to maintain the reactor power at its setpoint within the constraints that process systems can absorb this power. In the event of an upset, the regulating system has the capability of either slowly reducing reactor power (Setback) or, on all designs after Pickering A, of very rapidly reducing reactor power (Stepback) to match plant conditions.

The control computers are also given the job of annunciating some of the plant alarms, controlling turbine runup, controlling boiler pressure and reactor power distribution on all plants and additionally on some plants primary system pressure control, steam generator level control, moderator level control and fuelling machine control.

3.3.2 Electrical Power Supplies

Electrical power supplies are also automatically controlled so that they can be very reliable. The supplies are split into odd and even systems, either of which has the capability of operating the unit under shut-down conditions. Within each odd or even system, there is a hierarchical arrangement of distribution systems that supplies power to systems that can withstand various periods of power interruption. At the highest reliability are those systems that cannot withstand power interruption. These are supplied by batteries or inverters supplied by batteries. In general, these are instrumentation and control loads.

The next level of power supplies can withstand interruption for some period of time. These are normally supplied from the grid or from a service transformer on the unit's generator but are supplied from standby generators, driven by diesels or combustion turbines if the normal supplies are interruptued. It takes a few minutes for these standby supplies to come up to speed and voltage and to automatically pick up load. This level of supply can handle the station under shutdown conditions. This supply is also used to recharge the batteries as they are one of the loads that can be interrupted for only a finite time (the order of 40 minutes).

The lowest level in the power hierarchy comes from the grid or from the unit generator itself. These are required to operate the reactor at significant power as they drive the major plant equipment such as heat transport pumps, steam generator feed pumps, and condenser cooling water pumps. These systems are very reliable and have both automatic and manual provisions for obtaining power from various sources such as other busses of the same class or other units in the same station.

3.3.3 Operator Controls

Besides the automatic controls described above, there are manual controls, displays, alarms, and set point adjustments with which the operator interacts with the plant. The majority of these controls for each four unit station are located in a main control room, centrally positioned in the generating station. The controls are grouped in a logical fashion, by major plant system, on control panels which make up the walls of the control room.

The design of the control panels is carried out by a team of specialists in the field of human factors engineering and human-machine interface techniques. They employ such error reducing techniques as grouping equipment controls and indicators on mimic panels, standardized position indicators for valves, breakers, dampers etc. and annunciating important alarms with alarm windows. On more modern plants, extensive use of custom-designed CRT information display systems is employed to present to the operator the large amount of information entering the control room in a form that is most useful to him.

On the Pickering NGS B, Bruce NGS B and Darlington NGS, there is an additional seismically qualified control room, well separated from the main control room that contains sufficient controls and equipment to monitor and control the station under shutdown conditions, in the event that the main control room is uninhabitable or unusable.

3.4 Special Safety Systems

3.4.1 Introduction

To cater to process upsets or equipment failure that are beyond the capability of the process systems described above, all CANDU reactors have installed a number of additional systems, called "Special Safety Systems" that do not contribute at all toward the plant's main purpose of generating electricity. To the greatest extent possible these systems are independent of the normal process systems and from each other, especially in the realm of initiating signals. It is impractical to supply completely independent supplies of electrical power, instrument air, service water, etc, but attempts are made to minimize any interaction and to generally make the systems fail safe in the event of loss of instrumentation power or instrument air. For each reactor there are three types of special safety system:

a) Shutdown System
b) Containment System
c) Emergency Coolant Injection System.

In addition to these major system types, the latest designs also include a number of back-up process systems that improve the reliability of supply of electrical power and heat removal systems. These will be described further in Section 3.4.5.

3.4.2 Shutdown Systems

Incorporated in the design of all Ontario Hydro nuclear plants since Pickering NGS A, are two independent, diverse shutdown systems. Figure 3-1 shows a simplified representation of the two systems. Both systems have sensing circuits which monitor the status of a range of parameters and trip logic which initiates the shutdown action.

On SDS1 the reactor trip is effected by a set of neutron absorbing shutoff rods. These rods are normally held above the reactor by electro-magnetic clutches. On receipt of a trip signal (or loss of power) the clutches are de-energized and the SOR's drop quickly into the calandria by gravity. In later designs the rods receive additional initial acceleration from a compressed spring.

For SDS2 a liquid poison (gadolinium nitrate dissolved in D_2O) is injected into the moderator through a set of eight horizontally oriented injector nozzles. The liquid poison absorbs the neutrons required to maintain the fission process, thereby shutting down the reactor. The liquid poison is held in a set of tanks connected to a high pressure helium gas supply through a set of fast acting valves. On receipt of a trip signal or on loss of electrical power or instrument air, these valves are opened rapidly, pressurizing the liquid poison tank and forcing the poison into the moderator.

Important points to note are the diversity in the two methods of shutting down the reactor, the physical separation between the two systems, and their fail safe nature. The cabling, tubing and equipment associated with the sensing and trip logic systems are also physically separated. To the maximum practical extent, different equipment from different suppliers is used in the two systems where the same function is performed.

A number of plant variables are measured by each of the shutdown systems to determine when the system should be actuated. Sufficient variables are independently measured by each system that proper system operation can be reliably assured at all power levels and for all configurations allowed for normal operation. Where practicable two trip signals on different variables are provided on each shutdown system for each postulated event.

The process variables are combined in a two out of three majority vote logic which allows very reliable system operation while at the same time rejecting spurious operation of a single measurement. The majority vote scheme also allows testing of the instrumentation and logic during service which contributes to its very high, demonstrated reliability. If a failure is detected during testing, the particular logic channel can be placed into a safe state so that operation of either of the remaining logic channels will cause system actuation.

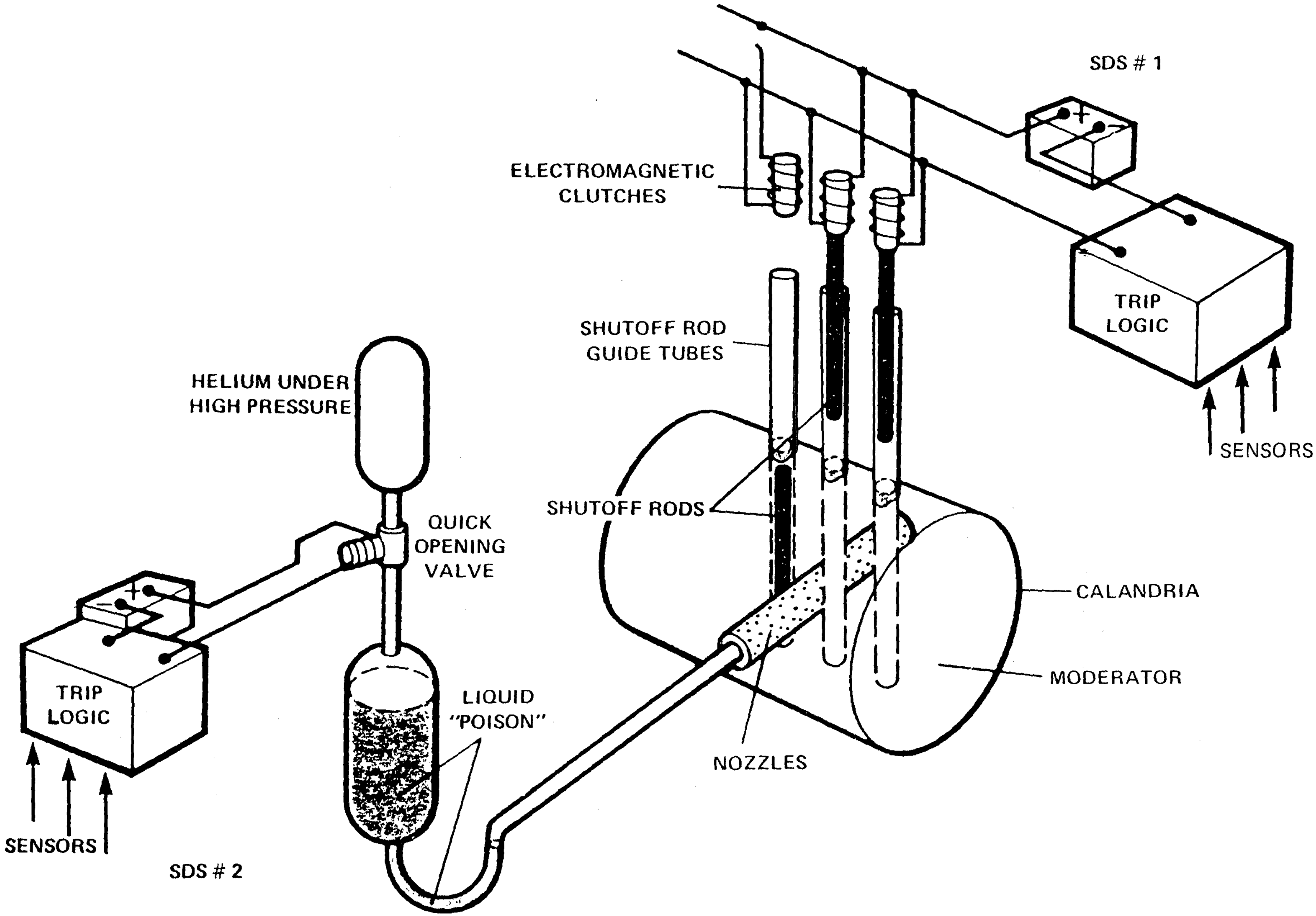

FIGURE 3-1
Reactor Shutdown Systems

3.4.2 Emergency Coolant Injection System

The emergency coolant injection system (ECIS) is designed to inject a reserved supply of light water into the heat transport system in the event that the loss of normal coolant cannot be made up by the normal process systems as described in Section 3.2 (a so-called "loss-of-coolant" accident or LOCA). A simplified schematic of the Bruce NGS B ECIS is shown in Figure 3-2. A similar system is installed on Bruce GS A. The system is made up of a high pressure supply of light water in the form of a set of accumulator water tanks and gas tanks and a lower pressure long term injection system involving a containment recovery sump, recovery pumps, storage tank and heat exchangers. These systems feed into a header connected to all four units. At each unit the header connects into the reactor headers. On Pickering NGS A and NGS B and Darlington NGS, high pressure pumps are used in place of the accumulator gas and water tanks.

On detection of a LOCA emergency coolant injection is initiated. The accumulator gas valves open along with the valves in the ECI supply lines to the accident unit. The HP gas drives the water from the accumulator water tanks into the headers of the accident unit. Part of this water passes directly out the break with the remainder passing through the reactor core before discharging from the break. In parallel with this initial injection the ECI recovery pumps are started. Once the accumulator storage tanks are depleted, water is pumped from the grade level storage tanks into the accident unit. In the long term, the coolant discharged from the break collects in a central containment recovery sump where it is recirculated through heat exchangers back into the accident unit.

The system operation just described is for the Bruce NGS configuration which uses compressed gas accumulators for the initial injection. On Pickering NGS B the initial injection uses electrically driven high pressure pumps which draw water from an elevated storage tank. When this water supply is used up, the supply for the high pressure pumps automatically switches to a set of lower pressure recovery pumps which draw coolant which has collected in the accident unit in recovery sumps. The recovered coolant passes through a heat exchanger before it is delivered to the high pressure pumps.

Pickering NGS A is in a state of transition. Its original design used the moderator system as a source of initial emergency coolant and then used the moderator pumps as well to re-inject the mixture of heavy water from the primary system and the moderator that collected under gravity in recovery sumps in the accident unit. The modified design will use the same source of high pressure initial injection as Pickering NGS B but will revert to the initial design's configuration for the recovery mode. The high pressure system is being installed on units 1 and 2 while they are out of service for retubing, and will be installed on units 3 and 4 in 1989 and 1988 respectively.

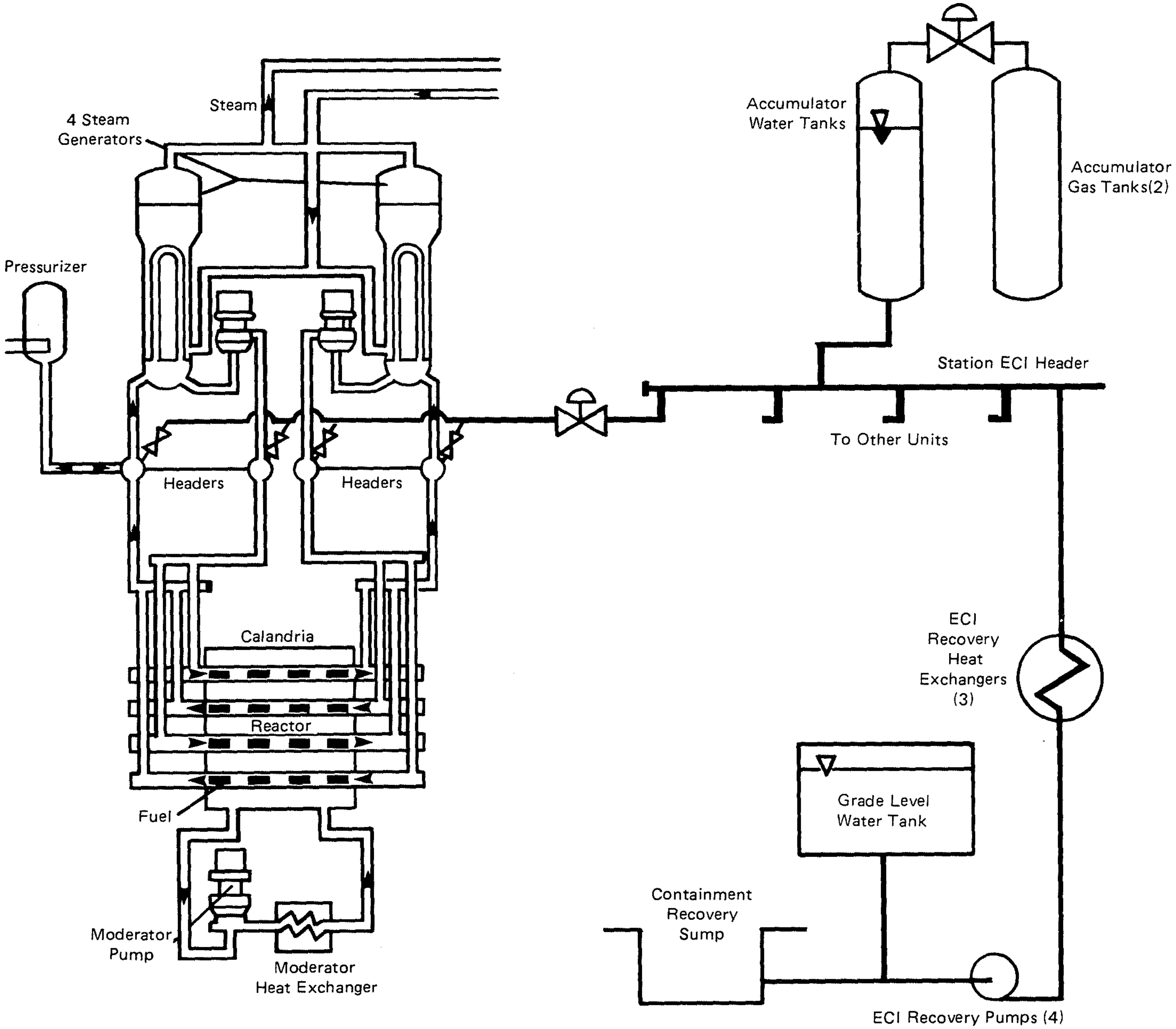

FIGURE 3-2
Emergency Coolant Injection System

In the Darlington NGS configuration, the initial injection also uses electrically driven high pressure pumps, but their suppy comes from low pressure pumps which in turn draw their supply from a grade level water storage tank. Upon depletion of the water stored in the injection water storage tank, a recovery mode (long-term injection) is established manually. During this phase, the low pressure pumps draw a mixture of natural water and heavy water from the ruptured heat transport system from a recovery sump in the pressure relief duct. A portion of the water discharged from the low pressure pumps may be returned to the pressure relief duct through heat exchangers to provide cooling of this supply of water.

Although the emergency coolant injection systems at each station differ somewhat, they are all effective at meeting their design objectives.

In a fashion similar to the shutdown systems, the ECI system also employs an independent multi-channel majority vote logic system that allows complete testing of the initiating and control logic while the system remains operational.

Besides redundancy in the instrumentation systems, the ECI system also has redundancy in its mechanical equipment. This redundancy allows full functional testing of the mechanical equipment such as isolating valves, pumps and heat exchangers.

The ECI systems have the capability of preventing fuel failures as a result of fuel overheating caused by a LOCA for all small breaks up to and exceeding a guillotine failure in the largest feeder. Beyond some break size we are currently unable to demonstrate that no fuel failures will results, but we can show that the injected coolant will eventually flood all channels and cool the fuel. By the time this "rewetting" occurs, some fuel sheaths may have failed - but releases to the public are well within regulatory guidelines for such low probability events.

3.4.3 Containment System

The final barrier to the release of radioactivity to the environment in the event of an accident is the containment system. This system ensures that only a minimal amount of radioactivity will escape in the remote possibility that some combination of events should result in a release of activity within its boundary.

The containment system provided for all Ontario Hydro's current and planned nuclear generating stations is of the vacuum containment concept. This concept is a unique Canadian design and is considered to be one of the best containment designs in existence today. In addition to the concept of vacuum containment, the system incorporates further notable design features which result in a system that is almost entirely self-actuating. That is, the major features of the system will respond as required without any external power requirements, or any need to signal or actuate active devices.

The containment system comprises the reactor building concrete envelope, a large concrete duct connecting the reactor building to a valve manifold containing several large valves in parallel, the valves themselves which separate the duct from the vacuum building, and the vacuum building. The large volume of the vacuum building is kept evacuated at an absolute pressure of about 10 kPa (1.5 psi absolute). The vacuum building also contains a self-actuating dousing system, which works on the syphon principle, which will cool and condense water and steam which enter the vacuum building. Figure 3-3 shows the main components of the containment system as well as illustrating how it works.

Within our various plant designs, there are two different arrangements of plant equipment with respect to the reactor building containment boundary. The original arrangement, used on Pickering NGS A and subsequently repeated on Pickering NGS B is shown in Figure 3-4. In this arrangement all of the heat transport, moderator and auxiliary systems as well as the steam generators are located within the containment structure. The containment boundary is formed by the external walls and roof of the Reactor Building. The Reactor Building containment volume is connected to the Vacuum Building (which is common to all 8 units at Pickering NGS) by the pressure relief duct at the boiler room floor elevation.

The containment design was modified for Bruce NGS A and subsequently repeated for Bruce NGS B and Darlington NGS. A cross-section of the Bruce NGS B arrangement is shown in Figure 3-5. The objective was to locate as much equipment as possible outside containment in order to facilitate maintenance in lower radiation fields, thereby reducing occupational exposure to radiation. This objective has been achieved as demonstrated by operating experience.

In this arrangement only the high pressure, high temperature portions of the heat transport system are located inside containment. Most of the heat transport auxiliary systems and all of the moderator and auxiliary systems are located in confinement rooms outside containment. One of the drawbacks with this design is the number of large penetrations through the containment boundary. There are penetrations at the heat transport pumps, steam generators and reactivity mechanisms deck. Considerable design effort has been applied to ensure that these penetrations will not be breached as a result of a large LOCA in adjacent piping.

To allow access into either containment configuration a number of airlocks are provided. A combined personal and equipment airlock is shown in Figure 3-6. The airlock is basically a cylindrical structure with a set of leaktight doors at each end. Each door is equipped with double inflatable rubber seals and a mechanical latching arrangement to ensure a leaktight seal. Interlocks are provided to ensure that only one set of doors can be opened at any one time.

During normal operation the pressure of portions of the containment with potential to contain radioactivity are held slightly subatmospheric to ensure inleakage, thereby eliminating any chronic release of activity.

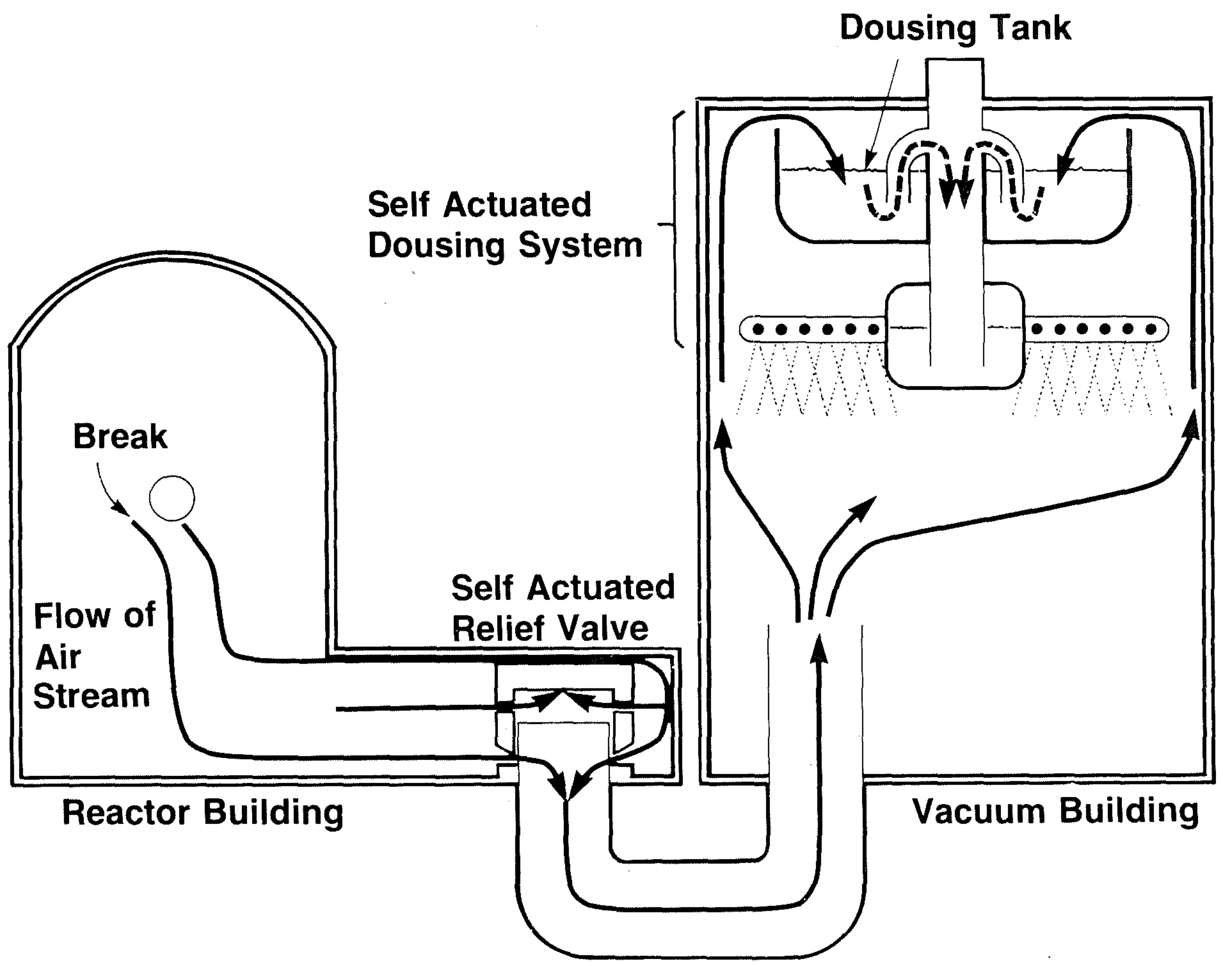

FIGURE 3-3
Operation — Negative Pressure Containment

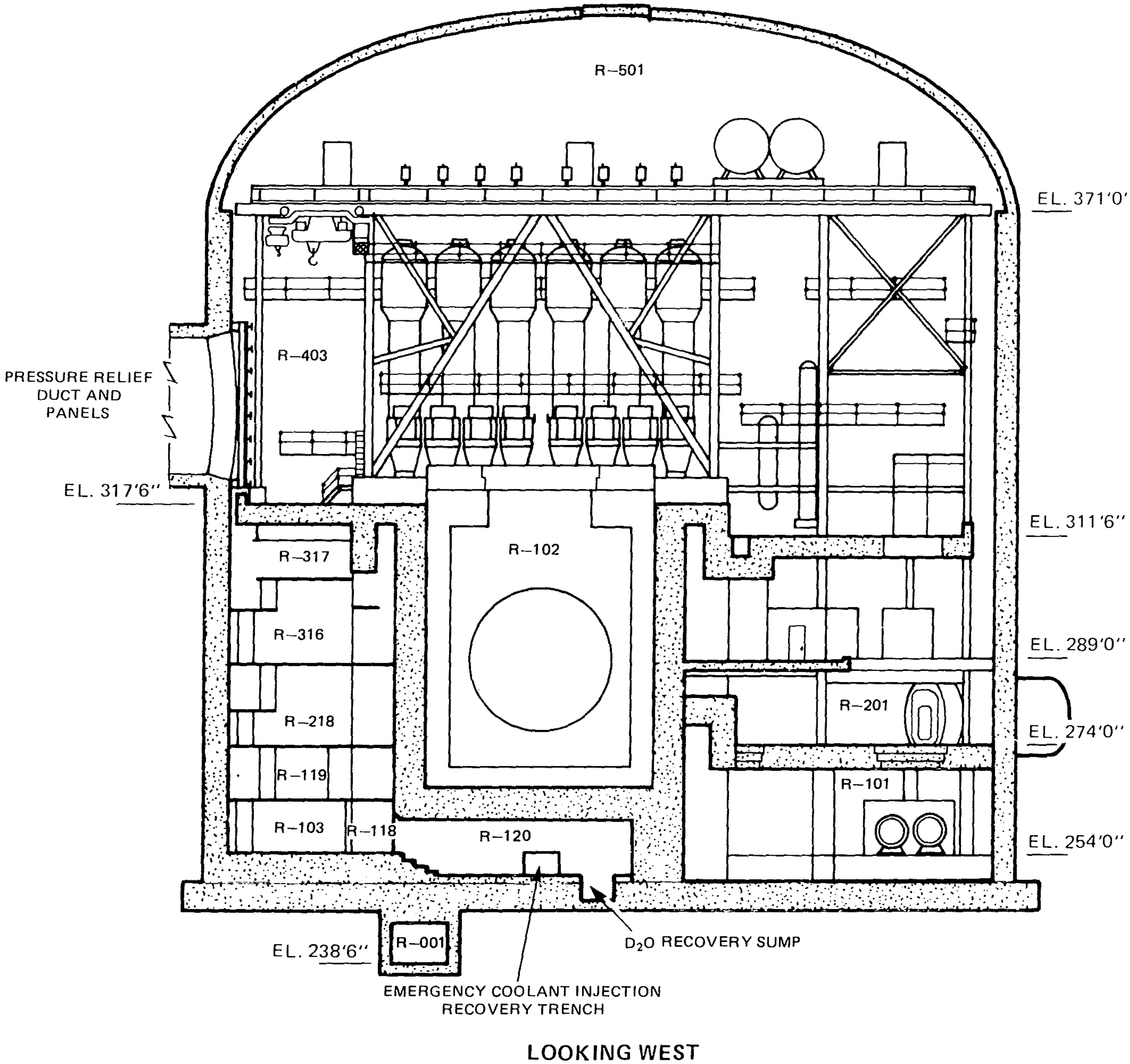

FIGURE 3-4
Pickering GS B Reactor Building Containment

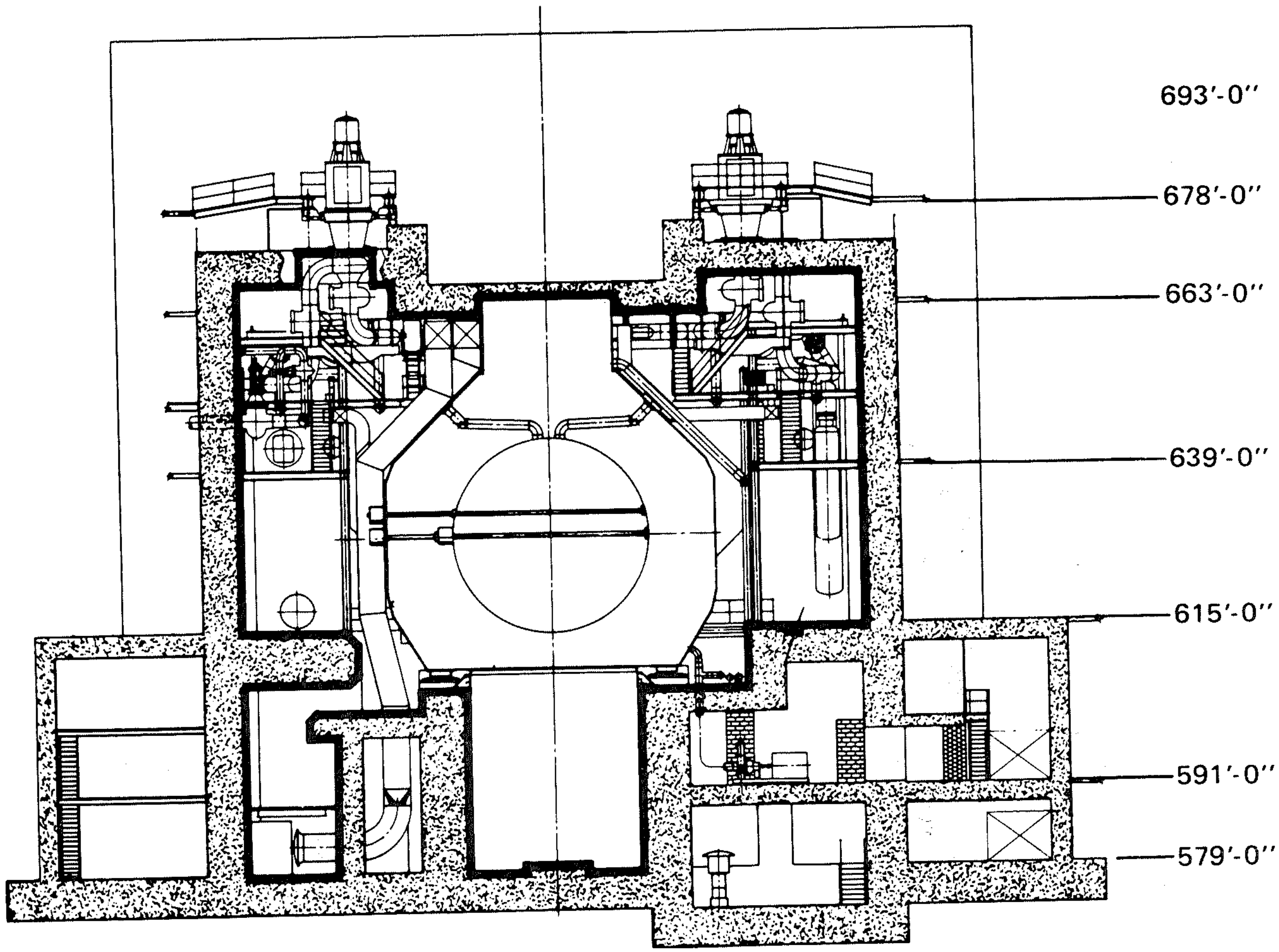

FIGURE 3-5
Bruce GS B Reactor Building Containment

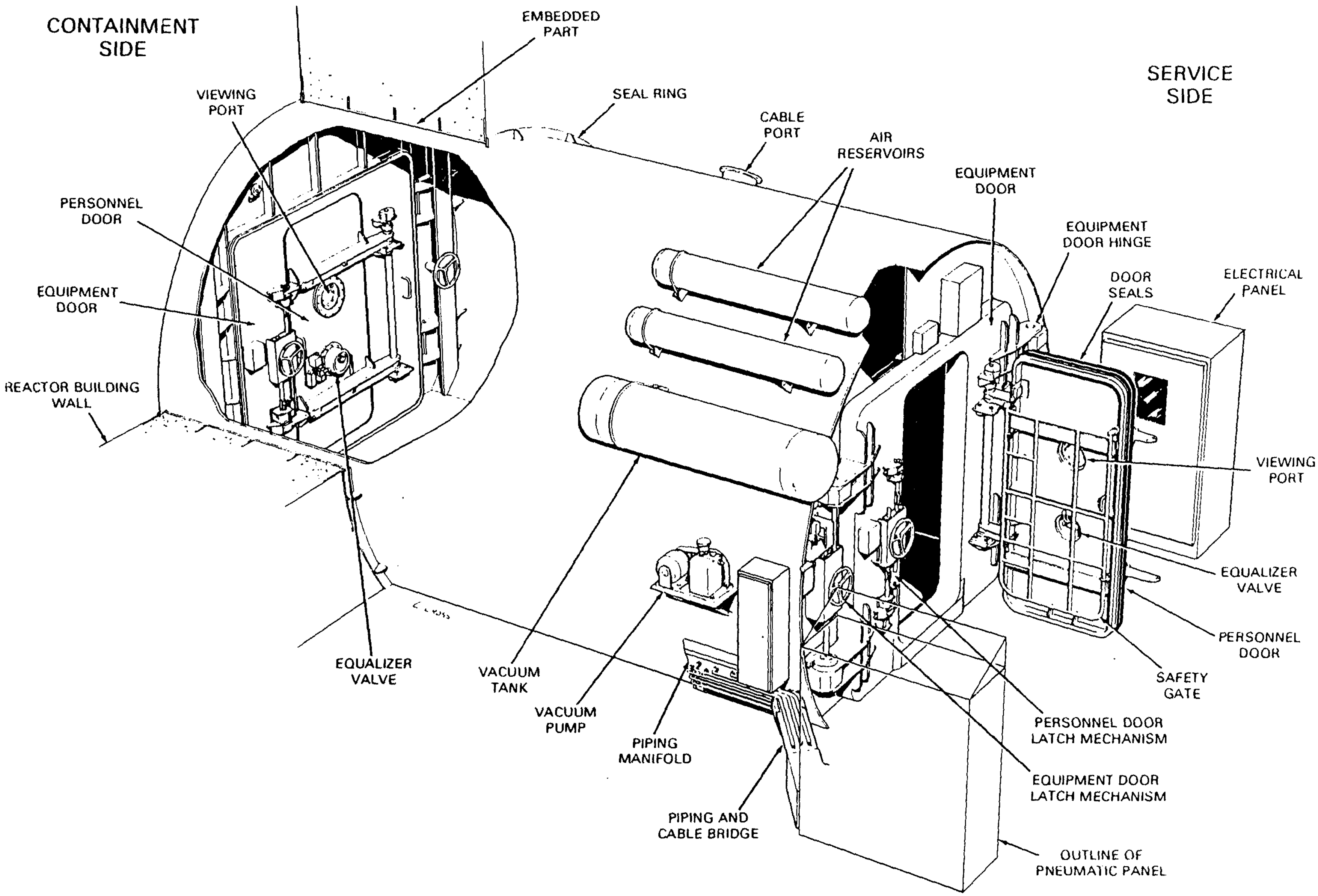

FIGURE 3-6
Typical Personnel and Equipment Airlock

Any air discharge required to maintain this pressure differential is exhausted via the active exhaust system. The effluent is filtered to remove particulates and iodine and the flow is monitored for activity. The flow is automatically stopped on a high activity signal within containment.

The design basis for the containment system is to hypothesize a massive failure in the reactor coolant piping circuit. This results in a release of high temperature steam into the reactor building and also leads to degradation of fuel cooling. This latter effect could result in a release of radioactivity from the fuel which would also enter containment.

The pressure rise in the reactor building, resulting from the release of high pressure steam from the hypothesized failure, acts to open the self actuating pressure relief valve valves which connect the reactor building volume to the vacuum building volume. The opening of these valves is accomplished by the actual pressure in the duct acting on the valves causing them to open; hence, the term "self-actuating".

Opening of these valves causes the air-steam mixture in the reactor building to be drawn into the vacuum building thereby rapidly returning the pressure in the reactor building to below atmospheric pressure. After that, any leakage in the containment boundary would cause air to enter the building; any radioactivity within the building could not escape. It should be noted that, because the period of overpressure is very small the escape of radioactivity would be negligible even if a significant "hole" in containment is assumed. In other words, the system will be very effective even with an impairment of the containment boundary. The major effect of an impairment in the containment envelope is to shorten the period before containment pressure increases to atmospheric pressure. For example, at Bruce NGS A, if the isolation dampers do not automatically close and no operator action is taken to close them, repressurization takes 19 hours versus 111 hours for an intact containment.

The air-steam mixture which passes into the vacuum building causes a rise in pressure which in turn initiates a large spray of cool dousing water from a tank contained in the top of the vacuum building. This condenses the steam and further reduces the pressure inside the containment structures. Again, this dousing spray is actually triggered by the pressure rise, and is, therefore, self-actuating.

Operation of the negative pressure containment system following a large LOCA is shown diagramatically in Figure 3-3. There is a large release of steam into the Reactor Building which results in a rapid increase in containment pressure. The self-actuated Pressure Relief Valves open automatically at 7 kPa (1 psig). The steam enters the Vacuum Building where the pressure increase actuates the dousing system. The resultant spray condenses the steam, thereby reducing the pressure. A typical pressure transient is shown in Figure 3-7. For the largest break size the pressure peaks within the Reactor Building accident vault in 3s and returns to subatmospheric in about 60s.

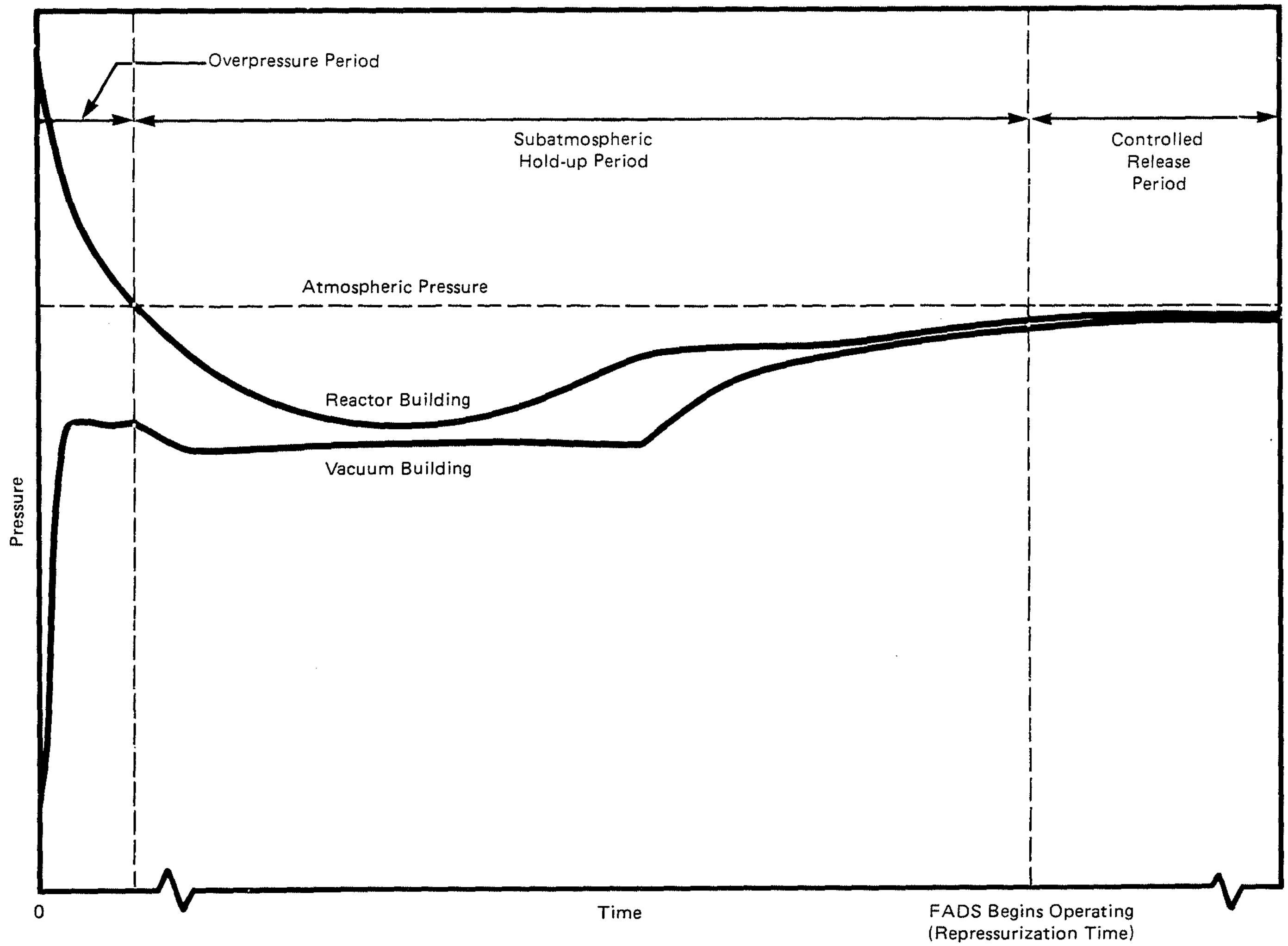

FIGURE 3-7
Typical Containment Pressure Response Following a LOCA

Assuming no containment impairments, the pressure slowly returns to atmospheric pressure as a result of the small inleakage of air from outside through the structures and because of the consumption of instrument air by equipment located within the containment.

When containment pressure returns to atmospheric, the Emergency Filtered Air Discharge System (EFADS) is brought into operation to control the pressure slightly sub-atmospheric. EFADS is equipped with particulate filters and charcoal filters for iodine removal, and radiation monitoring equipment to measure any radioactive releases from the station.

Hydrogen which can be produced by a Zircaloy/steam reaction during a LOCA is removed from the containment atmosphere by controlled ignition. The system utilizes a set of igniters located inside the Reactor Building vaults. The igniters are activated automatically on a containment "buttonup" signal. Ignition will occur when the post accident hydrogen concentration increases to the flammability limit and the humidity which suppresses ignition decreases.

3.4.4 Backup Cooling Systems

3.4.4.1 Emergency Water Supply

On Pickering NGS B, Bruce NGS B and Darlington NGS, there is a separate supply of service water to equipment that is required to operate to remove decay heat following an earthquake. This supply has a seismically qualified distribution system. Sufficient redundancy in supply pumps is provided such that the system can reliably supply all required loads. Each station has its own set of loads for this system, but typical examples of the loads supplied are:

- makeup to steam generators
- service water to vault coolers
- service water to ECI recovery heat exchangers
- makeup to the heat transport system.

3.4.4.2 Emergency Power Supply

The emergency power supply provides electrical power for equipment and instrumentation required to maintain and monitor the reactors in a safe shutdown state following a serious common mode event such as an earthquake or steamline break. One of the main loads for this system is the emergency water supply system described above.

The system consists of seismically qualified combustion turbine generator sets along with necessary distribution and control equipment.

The plant dynamics are such that the system can be a simple, manually operated and controlled system, with up to one-half hour available to put it into operation.

3.4.4.3 Steam Generator Emergency Cooling

The Steam Generator Emergency Cooling system is designed to provide an interim feedwater supply to the steam generators following the loss of the normal feedwater supply. For proper operation the steam generators must first be sufficiently depressurized as this system is designed to inject water at fairly low pressure. The steam generators and their control systems are designed so that they will always have sufficient water inventory to allow for this depressurization. The storage tanks are located at a high elevation so that water will flow by gravity to the steam generators. In some designs there is also an assist from pressurized air reservoirs. The capability of the storage tanks is sized so that the supply of make-up will be sufficient until long term supplies can be made available from the Emergency Water Supply described above.

3.4.4.4 Moderator System

A design feature of CANDU reactor cores is that the interstitial space between fuel channel assemblies is occupied by a completely separate, relatively cool, low pressure D_2O moderator. This fluid has two normal purposes; slowing down fast neutrons from the fission process to thermal velocities where they are again likely to cause fission of uranium atoms, and removing heat from reactivity devices, guide tubes, and calandria structures.

However, for certain loss-of-coolant accidents the moderator also maintains the fuel channel assemblies in a coolable geometry until the emergency coolant injection system is effective. In addition, if the ECI system should fail for some reason, the fuel channels will overheat and the pressure tube will sag into contact with the calandria tube, with fuel bundles slumped onto the bottom of the pressure tube. Heat transfer by conduction is sufficient to remove the decay heat so that fuel channel integrity is maintained. Thus the presence of the moderator virtually eliminates any concerns with the core melt accident.

3.5 Environmental Qualification

As a result of an accident occurring in the plant or as a result of some event occurring outside the plant (both natural and man-made), equipment may be subjected to other than its normal environment. If this equipment is credited with performing some preventative, mitigating or accommodating action, then there must be assurance that it will operate in the abnormal environment. Where the effects are pervasive, there is no option other than to harden or protect or isolate equipment that is credited from the environment. This is the approach taken to seismic events where all credited equipment is either type tested or demonstrated by analysis to perform as needed. Similarly, equipment in containment which is credited to perform after a LOCA is qualified to withstand the temperature, pressure, humidity and radiation environment that it would likely see. Aging effects are also considered.

For events where the conditions are less widespread, adequate separation of redundant equipment is a viable defence. For this approach, following a postulated event, sufficient equipment must be demonstrated to be unaffected by the event such that the necessary safety functions of

a) shutting down of the reactor
b) removing decay heat
c) monitoring the status of reactor and systems
d) monitoring for activity release

can be carried out reliably.

This is the philosophy behind the so called "two group" design approach where systems are allocated to either of two groups. Each group has the capability to carry out the above necessary safety functions. The goal is then to keep the groups sufficiently separated and isolated from each other that at least one would survive localized events.

Based upon the low probability of occurrance of initiating events, and the high probability of survival of equipment that is not formally environmentally qualified, it may be judged to be adequate to harden only one of the groups to environmental stresses.

This is the approach taken on Pickering NGS "B' and Bruce NGS "B" to design against earthquakes.

Safety Verification in Design

4.0 SAFETY VERIFICATION OF DESIGN

4.1 Introduction

The objective of the safety verification is to demonstrate that the design of the plant is such that the risk to the public is sufficiently low and that the regulatory requirements imposed by the AECB have been met. As discussed previously, the process involves several stages from early concept through commissioning of the plant prior to commercial operation.

During the definition phase of the project the conceptual design of both the plant and its safety systems evolve. Prior to receipt of a construction licence, sufficient safety analyses must be completed to demonstrate that the conceptual design has advanced to the point where assurance is provided that the final design will meet all safety requirements. Objective detailed design reviews of selected systems are undertaken by experienced designers not involved in the design. The design reviews are performed according to Quality Engineering procedures. Recommendations from this review must be addressed and suitably disposed by the project manager.

Throughout the acquisition phase of the project, as the detailed design is completed, further design reviews are conducted to assure design requirements have been met. The final safety analysis of the as built plant covers a wide range of accident scenarios and assessment of detailed system interactions which could not be considered at the construction licence stage.

Accident scenarios can be studied in a variety of ways - they can be analyzed with deterministic type rules or they can be examined using risk-based techniques. Over the years this latter process has evolved through the Safety Design Matrices developed for the Bruce and Pickering NGS B stations to the comprehensive Darlington Probabilistic Safety Evaluation (DPSE) undertaken for Darlington. The main objective of the DPSE is to provide a thorough safety design verification of the entire plant. Effects of all systems which might be called into action in the event of an accident are integrated. This process complements the individual system design reviews and ensures that the system interactions have been appropriately dealt with. Some accident scenarios, such as those involving common mode events, are more amenable to a deterministic approach where it is shown that sufficient equipment will be available to perform the required safety function.

As a final step prior to operation of the plant, the safety related systems are commissioned to ensure that their operation meets with the design intent. While it is obviously not possible to test them under accident conditions, special tests are devised and their performance under these conditions is analyzed.

The basic steps followed in this verification process are described in some detail in the following.

4.2 Accident Analysis

4.2.1 Purpose and Scope

Accident analysis is performed for two main purposes. The first one is to establish that the design requirements of those systems which have a bearing on nuclear safety have been met. This "design assist" analysis is carried out during the conceptual and early design phase of a project.

The other main purpose of accident analysis is to provide a demonstration that the radiological consequences of postulated system accidents are below the limits established by the Atomic Energy Control Board (AECB). This "licensing" analysis considers the final design or the "as built" plant.

The first step in accident analysis is the identification of the initiating events that can potentially lead to radiological consequences and their frequency of occurrence. This is accomplished through a comprehensive system-by-system review of the plant design, utilizing past experience with plants of similar design. Once the initiating events are identified, detailed simulations are carried out to predict the sequence of events and the response of the affected systems. Normally the simulations are terminated when a stable condition is re-established. At this point, the radioactive release from the plant, if any, is calculated and the resulting doses compared to the standards set by the AECB. Analyses are carried out assuming initial conditions pertinent to all anticipated operating states of the nuclear plants regardless of their frequency of occurrence. Examples include operation at all possible power levels and with some process equipment (pumps, valves, etc) not available. Furthermore, various degrees of impairment of the safety systems are postulated and analyzed.

The analyses carried out for licensing purposes generally employ assumptions and methods that lead to conservative predictions of the consequences of a given accident. However, all important physical phenomena are, where practical, represented in the mathematical models and sensitivity analyses are performed to estimate the consequences for realistic conditions. In addition realistic analyses are carried out to assist in establishing operating procedures for abnormal events. Accident analyses for operating nuclear plants are updated at intervals of approximately four years. However, the impact of experimental and analytical developments or changes in the performance of the plant's systems that could affect the conclusions of the safety analysis is assessed on an on-going basis.

Performing the accident analyses for a nuclear plant is a large task involving a variety of disciplines. The following subsections describe the typical accident classes that are considered and the analytical methods used in the analysis. The description of these methods is of necessity, brief. However, Appendix A provides a comprehensive bibliography for those readers interested in further details.

4.2.2 Postulated Accident Classes

A large number of initiating events leading to accidents can be postulated for a nuclear plant. Many of these accidents are common to other types of industry. Those unique to nuclear plants involve the potential for release of radioactivity.

These are discussed here.

A simplified representation of the causal mechanisms which could lead to release of radioactivity from the reactor core is shown in Fig. 4-1. Aside from the "common mode" events discussed in Section 4.3, six broad accident classes can be defined:

a) fuel and fuel handling failures
b) loss of electrical power supply
c) impairment of secondary side heat removal
d) loss of regulation
e) loss of coolant
f) failure of moderator pressure boundary

The safety target in accident analysis is to demonstrate that the radiological consequences to the public are acceptable. In reality, however, a set of derived targets is also used by Ontario Hydro. These derived targets provide more stringent requirements and thus are sufficient, although not necessary, conditions for ensuring that the radiological consequences are acceptably small. The use of stringent, derived targets serves two purposes. For certain accidents of relatively high frequency it ensures that the economic (as well as the radiological) consequences are minimized. For accidents of low frequency it avoids the need to deal with scenarios for which the consequences can only be speculative because of lack of suitable methods and/or experimental data. As an example, establishing the maintenance of fuel channel integrity avoids the need to analyze core disassembly scenarios. The various accident categories shown in Figure 4-1 and, where necessary, the derived safety targets are briefly discussed below.

4.2.2.1 Fuel and Fuel Handling Accidents

Fuel and fuel handling accidents are those which affect fuel bundles during some phase of the fuelling and storage process. Typical initiating events include loss of cooling to the fuelling machines and fuel mechanical damage either when the fuel is in the fuelling machines or in the irradiated fuel bay. These accident scenarios are analyzed to determine the timing and extent of fission product release and the corresponding doses to the public. The simulation of these events employs disciplines such as reactor physics, thermal hydraulics, fuel and containment behaviour.

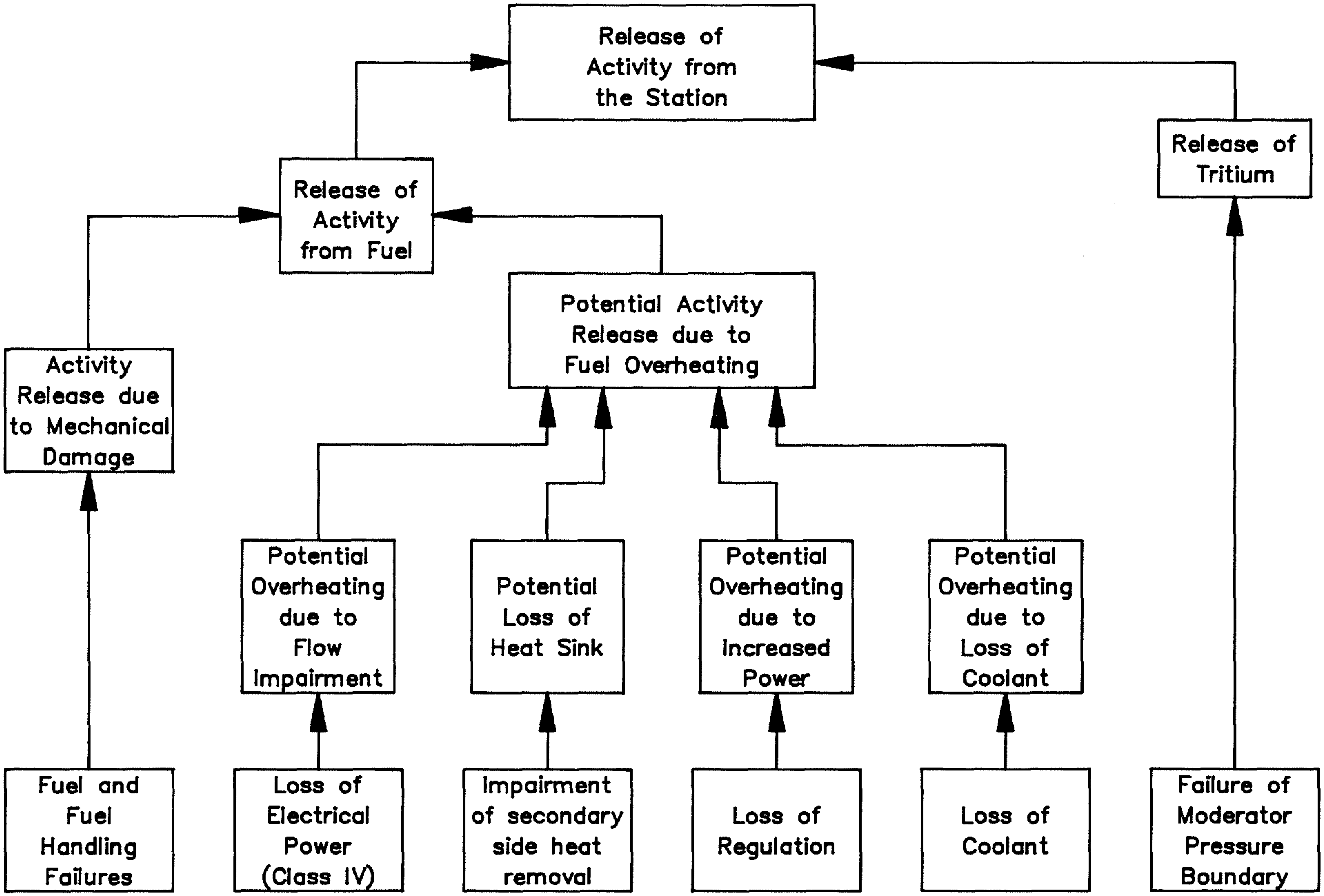

FIGURE 4–1
Potential Mechanisms for Release of Activity

4.2.2.2 Loss of Electrical Power

The electrical systems in a CANDU plant are grouped into four classes according to the interruptions of power that can be tolerated. Although partial and total failures of each class are analyzed, of particular interest is the failure of Class IV power since this leads to loss of forced coolant circulation in the reactor core and degraded heat sink capability.

The derived safety target for this type of event is prevention of damage to the heat transport system pressure boundary and prevention of fuel failures. The analysis is carried out by simulating, using suitable computer codes, the response of the heat transport and shutdown systems. It is verified that the shutdown systems are activated in time to prevent damage to the fuel and heat transport system which could potentially result from inadequate fuel heat removal or excessive heat transport pressure. It is also verified that in the longer term natural coolant circulation is adequate for fuel cooling and that a suitable alternate heat sink can be deployed.

4.2.2.3 Impairment of Secondary Side Heat Removal

Failures of the steam supply system or the steam generator feedwater system lead to a degradation of the normal heat sink for the heat transport system. This could lead to inadequate heat removal from the fuel and excessive heat transport pressure. The derived safety target for this type of event is to prevent fuel temperature excursions and damage to the heat transport system pressure boundary. The analysis verifies that a timely reactor shutdown takes place and that the remaining inventory in the steam generator is such that the operator has sufficient time to establish an alternate heat sink.

Thermal hydraulic computer codes are used to simulate a variety of initiating events such as breaks of various sizes in the steam and feedwater piping. The possible detrimental effect of steam on the plant equipment is also considered.

4.2.2.4 Loss of Regulation

Regulation of a CANDU reactor unit is carried out by means of a computer system designed to a high level of reliability. However, failures of either the computers or their external logic may result in the loss of control functions such as reactor power, heat transport pressure or inventory control.

Some of these events could result in a mismatch between the fuel heat generation rate and the rate at which it is transferred to the coolant with consequent fuel temperature and heat transport system pressure excursions. The analysis verifies that the shutdown system design is adequate to terminate the transient in time to avoid fuel and heat transport system damage. This type of event therefore does not lead to release of radioactivity into containment and emergency coolant injection is not required.

4.2.2.5 Loss of Coolant Accidents (LOCA)

Heat transport system piping failures call into play all the major safety functions such as reactor shutdown, emergency core cooling and containment. They are, therefore, used to set the requirements for the design of special safety systems. Initiating events considered in the analysis include breaks of all possible sizes up to a guillotine-type break of the largest diameter piping in the heat transport system, as well as all possible pipe locations. Loss of coolant accidents are usually classified as "small" or "large" depending on whether they are initiated by ruptures of small-diameter piping (typically less than 8 cm in diameter) or large diameter piping. The former constitutes approximately 99 percent of the total heat transport system piping. A "small" break is, therefore, relatively more probable than a "large" break. Recognizing this, the derived safety targets are more stringent for the case of a "small" loss of coolant accident.

For a "small" LOCA the analysis demonstrates that the safety features (shutdown and emergency core cooling) are such that no fuel dryout is reached prior to reactor shutdown and no fuel failure due to overheating occurs thereafter. Achieving this minimizes the economic as well as the radiological consequences of the accidents. For the low probability, "large" loss of coolant accidents, fuel damage and release of radioactivity into containment cannot be precluded. For this type of accident, the analysis determines that the dose limits established by the AECB are met. However, the derived safety target of maintaining fuel channel integrity is imposed. Achieving this target ensures that the fuel remains in a "coolable" geometry throughout the transient and that the long term emergency coolant is capable of removing the decay heat. This implies that regimes which would make the analysis highly speculative because of incomplete technical knowledge, such as the phenomena associated with core disassembly, are avoided.

4.2.2.6 Failure of Moderator System Pressure Boundary

A piping failure in the moderator system leads to a decrease in the moderator inventory in the calandria. Moderator draining does not disturb the coolant flow in the fuel channels and, if allowed to continue, leads to a reactor shutdown. This type of accident, therefore, does not lead to activity release from the fuel. After operation however, the moderator contains a significant amount of the radioactive isotope tritium. The radiological consequences of releasing tritium are evaluated in safety analysis.

4.2.3 Analytical Methods

As apparent from the brief description of the accidents considered in safety analysis, several disciplines are required to perform the analyses. The methods used by Ontario Hydro, within each discipline, are generally state-of-the-art in the industry. They have been developed over a number of years and are continuously updated to include new experimental results or advances in analytical and computational techniques. Verification and further development of these methods is supported by a large experimental program conducted both within Canada and internationally in partnership with various organization such as AECL and EPRI in the USA.

The analytical steps required to carry out the analysis of loss of coolant accidents are shown in Figure 4-2. At least some of the same steps are also necessary for the analysis of the other accident categories. A brief description of the methods currently used to carry out each step is given in the following.

4.2.3.1 Reactor Physics

Reactor physics is concerned with the fission chain reaction and, specifically, with methods of maintaining a reactor critical with the desired spatial distribution of fission rates. In accident analysis, the role of reactor physics is to provide computations of the core power distribution prior to the accident and of the changes taking place during the accident.

To establish the initial power distribution, static reactor calculations are carried out. The basic procedure followed (Ref. 1.1, 1.2, Appendix A) is to first derive a set of cross-sections for the "basic" cell assumed to be part of an infinite array of such cells. This basic cell consists of a portion of a fuel channel surrounded by an appropriate volume of heavy water moderator. The computer code POWDERPUFS is used for production simulations while the WIMS and LATREP codes are used to verify the validity of the POWDERPUFS prediction for unusual conditions.

Changes to the cross sections of the basic cell, due for example to the presence of control devices, are determined separately using three-dimensional neutron flux calculations performed over a core volume ("supercell") containing the materials present in the basic cell and the control device. Two energy group cross sections are used since experience has shown that dividing the neutron energy spectrum in two discrete groups gives sufficiently accurate results. Once cell-averaged cross sections are obtained, the next step is to solve the space energy equations for the neutron flux over the reactor core. Computer programs such as OHRFSP (Reference 1.9, Appendix A) are used. The diffusion rather than the more rigorous transport equation is adequate for these calculations. The combination of these two approximations, two energy groups and diffusion theory, make it possible to perform calculations in three-dimensions at a reasonable computer cost. Various methods are used

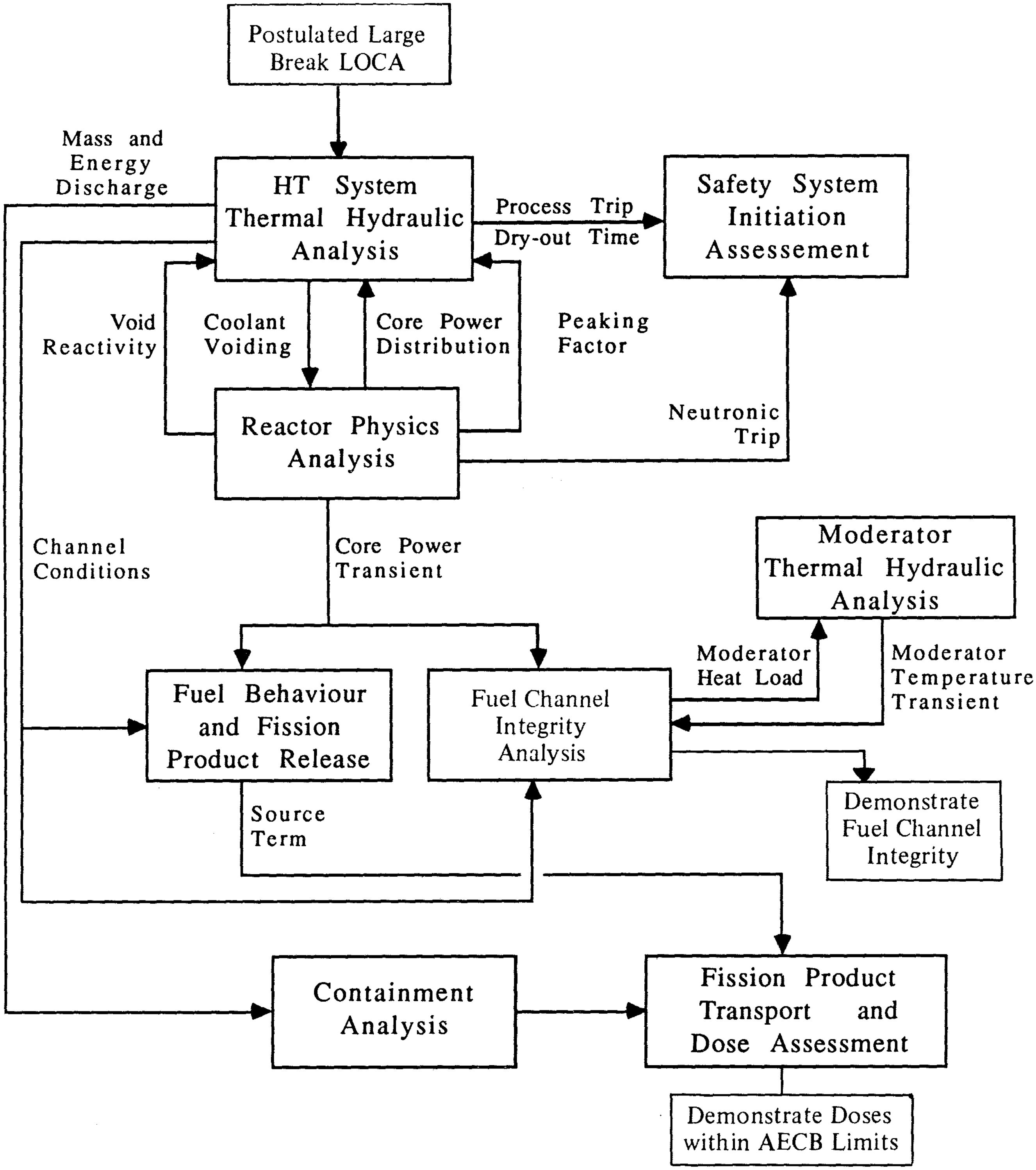

FIGURE 4-2
Basic Modules Used in LOCA Analysis

to compute the parameters of interest to accident analysis. These methods include those needed to derive the core power distribution as it exists "on average" over a long period of reactor operation as well as "instantaneous" distributions corresponding to some "instant" in the irradiation history of the reactor core.

For dynamic reactor calculations a hierarchy of methods of varying complexity and spatial detail is used (Ref. 1.4, 1.6, 1.7, 1.8, Appendix A). These are point Kinetics methods for inexpensive survey type calculations, three dimensional modal expansion methods for "production" analysis and a three dimensional code based on the Improved Quasi Static method (Ref. 1.5, Appendix A) for accurate, reference calculations.

A significant effort has been devoted to verifying the computer codes against data obtained from power and research reactors. In some instances, where it is not possible to validate a code against experimental data, (i.e. large loss-of-coolant accident), inter-code comparisons are used for validation. Reference 1.11 provides a description of some of the validation studies performed.

4.2.3.2 Thermal-Hydraulics

The role of thermal-hydraulics in accident analysis is to calculate the response of the reactor systems affected by the postulated events. These systems include the heat transport system, the emergency coolant injection system and the steam and feedwater system. Results of thermal-hydraulic analyses include the timing of safety system initiation and transient behaviour of parameters such as coolant pressure, temperature and void fraction. This information provides boundary conditions for the detailed assessment of safety system effectiveness, adequacy of heat sink provisions, fuel and containment behaviour.

The main tool used for the simulation of the heat transport system response is the SOPHT computer code (References 2.1, 2.2, 2.3, 2.4 - Appendix A). This code contains detailed models of the piping and components of the specific reactor system as well as models of the control systems. The piping, fuel channels, pumps, and other components that comprise the various systems are represented by a series of linked control volumes. The flow pattern is assumed to be one-dimensional, and the fluids are assumed to be in an homogeneous equilibrium state within a control volume. Equations defining conservation of mass and energy are then applied to each control volume, and an equation defining conservation of momentum to each link. The resulting system of equations is simplified and solved using an implicit algorithm.

Comparison with experimental data has shown that this modelling approach gives sufficiently accurate results for most accident scenarios and, in particular, that the bulk system thermal-hydraulic behaviour (heat transport system pressure, inventory transient) is adequately predicted even for very severe conditions such as those encountered in a large break loss of coolant accident (References 2.5, 2.6, 2.7 - Appendix A). It is recognized, however, that the fuel and fuel channel behaviour during certain accidents is governed by local rather than bulk thermal-hydraulic conditions. In cases where separation between the vapour and the liquid phases is likely the code's predictions are not utilized directly and recourse is made to a conservative limit-consequence approach (Reference 2.8 - Appendix A).

In addition to the models used to determine the system response, a number of thermal-hydraulic methods are employed to address specific aspects of the accident analysis. The salient features of these methods are described below:

(a) Simulation of Individual Fuel Channels

In the SOPHT model, the fuel channels in the reactor core are usually represented by one "average" channel. While this is adequate for assessing bulk system response, certain accident scenarios require detailed thermal-hydraulic conditions for individual channels. Examples are the behaviour of the channel having the maximum fuel power or of the channels containing special instrumentation.

A computer code called MINISOPHT (a simplified version of the SOPHT code) is used to determine the coolant conditions and the heat transfer mode from the fuel to the coolant for the fuel channel of interest. MINISOPHT models the geometry of a specific fuel channel between the reactor headers and uses the coolant conditions in the headers as predicted by the SOPHT computer codes as boundary conditions.

For accidents such as small loss of coolant coincident with unavailability of emergency coolant injection, conditions of reduced pump head and low coolant subcooling may lead to coolant stratification in some fuel channels. Under these conditions, fuel elements and pressure tube surfaces exposed to steam cooling gradually heat up at a rate governed by the coolant liquid available in the feeders and headers and, eventually within the channel itself. A computer code named SLLOH has been developed by Ontario Hydro to simulate the response of a fuel channel to convective steam cooling (Reference 2.9 - Appendix A). The code accounts for steam generation within the channel, draining of the feeder and boil-off of the remaining coolant inventory in the channel. A detailed model of the fuel bundle is used to calculate the temperature of the fuel sheath and pressure tube.

(b) Simulation of Loss of Forced Circulation Events

A number of postulated accident scenarios lead to a loss of forced circulation in the heat transport system due, either, to an assumed, or consequential loss of electrical supply to the heat transport pumps. In this event, single-phase or two-phase thermosyphoning develops and maintains adequate fuel cooling after pump rundown. Given the differences in individual channel characteristics, however, there are situations where the effectiveness of thermosyphoning may deteriorate in some fuel channels. To determine the fuel and pressure tube temperature for these cases, a special methodology termed CCAFF (Channel Cooling in the Absence of Forced Flow) has been developed (References 2.10, 2.11, 2.12 2.13, 2.14 and 2.15 - Appendix A). This methodology considers various modes of natural flow circulation and, based on the system conditions, identifies the one that predominates. The methodology is based in part on experimental results obtained in the Cold Water Injection Test (CWIT) facility at Westinghouse Canada Inc., in Hamilton.

(c) Simulation of Steam Generator Blowdown

Although the SOPHT model contains a full representation of the secondary side system, the model is not sufficiently detailed to calculate the response of the steam generators to a break in the steam supply system. Depending on the severity of the break, the rapid depressurization and depletion of the steam generator water inventory may require an early reactor shutdown or other measures. The response of the steam generators to these accidents is simulated with the STGEN computer code (References 2.16, 2.17 and 2.18 - Appendix A). This code calculates parameters such as the discharge flow, the liquid inventory remaining in the steam generator as a function of time or the water level transient. Knowledge of all of these parameters are necessary to verify the availability of adequate heat sinks for the heat transport system. STGEN can also be used to predict the discharge from a pressurized vessel, as for example, in the case of a break at the top of the pressurizer (Reference 2.19 - Appendix A).

(d) Simulation of the Moderator System

As discussed previously, maintenance of fuel channel integrity is a safety objective. For some postulated accidents, the moderator is required to act as a heat sink in order to verify that this objective is met. The adequacy of this heat sink is demonstrated by comparing the subcooling required to avoid sustained calandria tube dryout (Sec. 4.2.3.3) with the subcooling available. A computer code named MODTURC has been developed by Ontario Hydro to calculate the steady state and transient temperature distribution in the moderator system and thus the subcooling available (References 2.20, 2.21, 2.22 and 2.23 - Appendix A). The code employs an explicit finite difference algorithm to solve conservation equations for mass, momentum, energy and dissipation rate of turbulence. The equations are expressed for three dimensional flow and formulated in cylindrical coordinates. The code's predictions have been verified with data obtained in special tests carried out in the Bruce NGS A and Pickering NGS B reactors (References 2.24 and 2.25 - Appendix A).

4.2.3.3 Fuel and Fuel Channel Analysis

Fuel and fuel channel analysis is performed to derive the transient fission product release into containment, which, in turn, is used to evaluate the radiological consequences of a postulated accident. Following irradiation, a large variety of fission products reside within the fuel elements. The fission products have different physical and chemical characteristics. For instance, their half-lives range from seconds to thousand of years, some are chemically inert while others are very reactive. These characteristics influence the radiological significance of a particular species upon release into the environment. Short-lived volatile species, producing intense gamma fields (Xe-135) act as contributors to external dose. On the other hand, chemically reactive species with intermediate half-life (I-131) can accumulate in certain organs of the human body and thus act as a source of internal irradiation upon inhalation or ingestion of radioactive material. These characteristics also affect the transient release of fission products. Volatile inert gases are most easily released in accident situations while species with high boiling points are retained in the fuel for all but the most severe accident scenarios.

The fuel inventory of fission products and its isotopic composition are functions of the fuel power irradiation history and are calculated using the ORIGEN computer code (Reference 3.1 - Appendix A). This is a reference code widely used in the nuclear industry and has been extensively verified.

During normal operation, the fission products reside in the fuel UO_2 grains, in closed cavities at the grain boundaries and in open cavities such as the gap between the Zircaloy sheath and the fuel pellet (References 3.2 - 3.5 - Appendix A). While the "gap" inventory of fission products can be easily released upon sheath failure, the "grain" inventory can only be released at elevated temperatures, well above normal operating temperatures, or if the fuel is subjected to severe fragmentation (References 3.6 - 3.16 - Appendix A). Therefore, the rate of fission product release depends also upon the inventory distribution within the fuel element. The distribution is calculated using the CURIES-II computer code (References 3.5, 3.17 - Appendix A), which is partly based on a widely used CANDU fuel performance code named ELESIM (References 3.2, 3.18 - Appendix A).

The extent and timing of fuel sheath failures following an accident is determined by the thermal and mechanical behaviour of the fuel and fuel channel (References 3.19 - 3.25 - Appendix A). A number of analytical methods are employed depending upon the severity of the transient. If the fuel temperature excursions are such that no significant fuel and pressure tube geometrical changes occur, the computer codes HOTSPOT (Reference 3.26 - Appendix A) and ELOCA (References 3.27, 3.28 - Appendix A) are used. If the mismatch between power generation and heat removal is severe, geometrical changes in the fuel (References 3.29 - 3.35 - Appendix A) and pressure tubes take place and are enhanced by the exothermic reaction between the fuel sheath material (Zircaloy-4) and steam (References 3.36 - 3.42 - Appendix A). These geometrical changes include radial strain of the fuel sheath (ballooning), fuel element sagging into contact with each other and relocation of molten Zircaloy (References 3.43 - 3.45 - Appendix A). Similarly the pressure tube may sag or strain radially to contact the calandria tube (References 3.46 - 3.59 - Appendix A).

To simulate the behaviour of a fuel channel under severely degraded cooling conditions, the CHAN-II (References 3.60 - 3.62 - Appendix A) computer code is employed. In this code each fuel bundle is represented as a series of concentric rings with coolant flowing in annular subchannels. Flow intermixing is modelled at fuel bundle junctions. The feedback due to geometrical changes is explicitly taken into account. Because of the broad range of possible geometries, parametric analyses are carried out and the fuel temperature transients used in estimating the fission product release are chosen conservatively to envelope the possible range of temperature excursions.

Once the temperature transients are calculated, the resulting fission product release is calculated using the CURIES-II computer code. This computer code includes models to represent all the known high temperature fission product release mechanisms such as diffusion of atoms in UO_2, oxidation enhanced grain boundary motion and fission gas sweeping, and liquefaction of the fuel matrix upon alloying with molten Zircaloy. The evaluation of the transient fission product release assumes that fuel channel integrity is maintained during the accident. As indicated previously, the maintenance of fuel channel integrity is a convenient derived safety design target which makes the analysis tractable and well supported by experimental data (References 3.63 - 3.67 - Appendix A).

A separate fuel channel behaviour analysis is performed to verify that this safety target is achieved. When the pressure tube is subjected to elevated temperatures it deforms plastically by a combination of circumferential stress (due to internal pressure) and lateral stress (due to gravity loading). Experiments show that such deformation can lead to contact between the pressure tube and the calandria tube and to a spike in the heat flux to the moderator surrounding the calandria tube (References 3.68 - 3.73 - Appendix A).

The potential for further deformation depends on whether this spike in heat flux is sufficient to exceed the critical heat flux (break down of the fluid film) on the calandria tube surface. If dryout does not occur, the pressure tube regains its strength as it cools down and no further deformation takes place. Essentially the analysis of channel integrity following contact between the pressure tube and the calandria tube is reduced to demonstrating prevention of sustained calandria tube dryout. The computer code (CHAN-II) used for fuel behaviour analysis is also used here. However assumptions and thermal hydraulic boundary conditions are selected to provide a conservative assessment from the standpoint of potential channel failure. In this assessment, consideration is also given to the potential for pressure tube failure before contact with the calandria tube. Potential failure mechanisms such as non-uniform radial strain due to localized hot spots are also evaluated.

4.2.3.4 Containment Analysis

The containment comprises systems and sub-systems whose collective function is to mitigate the consequences of accidents that result in a fission product release from the fuel, such that regulatory dose limits are not exceeded. To verify the adequacy of the containment design, analyses of the transient thermal hydraulic response and fission product behaviour within the containment envelope are performed. Release of fission products from the containment are calculated and used to predict potential radiological consequences to members of the public.

The containment response to the majority of accidents can be divided into three periods: an initial over-pressure period, a period of subatmospheric operation and a prolonged period of controlled discharge. The initial over-pressurization is caused by flashing to steam of the high enthalpy coolant discharged from the break. Venting into the vacuum building combined with cold water dousing is sufficient to return the pressure to subatmospheric levels. In the second period, the pressure decreases initially due to steam condensation but eventually begins to rise due to inevitable, small amount of air in-leakage. In the third period, the Emergency Filtered Air Discharge System (EFADS) is activated to maintain the containment pressure slightly below atmospheric and provide a controlled filtered discharge path for the radioactivity.

The transient thermal hydraulic response such as pressure and temperature distribution and fission product transport in containment in the short term is calculated using the PATRIC (Pressure and Temperature Response in Containment) computer code, which has been developed by Ontario Hydro. The containment system is represented as a number of control volumes called nodes. The fluid flow path (link) is assumed to be one-dimensional between nodes. Equations of conservation of mass, energy and momentum for a two component system of air and steam are advanced in time by a semi-implicit reduction scheme. The air and steam are assumed to be in thermal equilibrium within a node and travel at the same velocity between nodes.

PATRIC includes models of the containment isolation system, the containment pressure suppression systems (including heat sinks), Pressure Relief Valves (both instrumented and passive) and dousing, and has the ability to model different containment states. The PATRIC code has been verified against various fundamental problems, commissioning results and containment experiments (References 4.3 and 4.4 - Appendix A). The

PATRIC code uses the results of the thermal-hydraulic analysis and fuel release analysis as inputs to simulate the release of fission products to the environment during the short term over-pressure period.

Analyses are also performed to confirm that the integrity of the containment system is not impaired as a result of an accident (Reference 4.5 - Appendix A). The PATRIC code is used to determine the short term pressure and temperature response of the containment to loss of coolant accidents in the Heat Transport System and the Steam Supply System. The VENT code is used to simulate the containment response following combustion of hydrogen which may be produced by oxidation of the Zircaloy fuel sheaths.

Fission product transport and decay within the containment during the sub-atmospheric hold-up period is modelled using the FISSCON-II computer code (References 4.6 and 4.7 - Appendix A). FISSCON-II models deposition and desorption of iodine species for structures and water volumes in containment, transport of fission products between the main containment volume and the vacuum structure and decay of fission products within the containment. This code was developed by Ontario Hydro and has been verified against experimental data and the CORRAL-II code developed for the USNRC Reactor Safety Study (WASH-1400) (Reference 4.9 - Appendix A).

The releases to the environment as a consequence of controlled venting are also calculated using the FISSCON-II code which includes a model of the EFAD system simulating the iodine trapping efficiency and desorption from the filters.

Off-Site Consequence Analysis

The radiological consequences of fission product release are a function of the magnitude and nature of the release, the appropriate Dose Conversion Factors (Reference 4.10 - Appendix A) for the nuclides released and the atmospheric conditions at the time of the release. The radiological consequences to the public of this release are determined for both a member of the most radiologically sensitive age group assumed to remain at the site boundary for the duration of the accident, and the population within a 100 km radius of the accident. The specific dose pathways modelled in the dose calculation are external irradiation due to "immersion in" or "exposure to" a cloud of airborne material and internal irradiation to inhalation of material in the cloud.

Atmospheric dispersion is a function of the assumed meteorological and wind conditions and is modelled for three separate periods (Recommended in Reference 4.11 - Appendix A):

1. A short term release period lasting up to one hour during which the weather conditions and wind direction do not change (References 4.12 and 4.13 - Appendix A).

2. A prolonged release period lasting from one hour to one day during which the main wind direction is assumed to remain constant with some meander about this direction and the atmospheric stability conditions may change.

3. A long term release period during which both the meteorological conditions and wind direction may change.

For short term releases, Pasquill Stability Category F is assumed with a wind speed of 2.0 m/s in the direction of the critical individual. During the prolonged release period category F stability conditions are assumed to exist 50% of the time and stability category D conditions are assumed to exist for 50% of the time. Analysis of hourly weather data indicates the probability of these weather conditions actually occurring is less than 10%. For analysis of the long term release, site specific wind speed and stability information is used in the dose assessment.

Population distributions are projected to the predicted end of station life using the latest available statistics, and from these population distributions the population densities are calculated for varying band boundaries within 22.5° sectors. The site-specific population densities are used to calculate populations doses.

4.2.4 Experimental Support

A research and development (R&D) program to develop a better understanding of the phenomena of interest to reactor safety, and to provide a data base to develop and validate the computer codes used in accident analysis, has been under way for many years. The work is largely carried out at the AECL laboratories and funded by a consortium comprising Ontario Hydro, Hydro Quebec, New Brunswick Electric Power Commission and AECL. The budget is currently in the order of $15 M/annum. The program is divided into the following broad areas:

A. Fuel Channel High Temperature Transient
B. Fuel High Temperature Transient
C. Containment
D. Calandria Tube Integrity
E. Safety Thermal-hydraulics
F. Moderator Circulation
G. Fuel Channel Critical Power

Technical guidance in each area is provided by a Working Party comprised of technical experts from each organization. The working party periodically reviews the progress to ensure that high priority concerns are adequately addressed and new proposals are incorporated into the overall program.

The following provides a brief overview of the objectives of each main area.

A. Fuel Channel High Temperature Transient Program

The objective of this program is to investigate the behaviour of the fuel channel under severely degraded fuel cooling conditions. Under these conditions, the pressure tube may deform into contact with the calandria tube and thus establish a heat transfer path from the fuel to the moderator. A considerable effort was devoted in the past to studying the moderator subcooling required to maintain channel integrity for uniform contact between the pressure tube and the calandria tube. Currently the effort is concentrated on investigating the effect of pressure tube non-uniform circumferential temperature distributions as well as on tests needed to provide further verification of the accident analysis code CHAN-II.

B. Fuel High Temperature Transient Program

This program studies the behaviour of the fuel under the elevated temperature conditions predicted in severe postulated accidents. Issues of importance are the transient deformation of the fuel bundle, the failure of the sheath, the interaction of molten Zircaloy with UO_2, the fission product release characteristics, and the transport and retention of fission products in the heat transport system piping. A large experimental facility named BTF (Blowdown Test Facility), which will allow in-reactor fuel testing at temperatures in excess of 2000°C, is being commissioned at Chalk River Nuclear Laboratories (CRNL).

C. Containment Program

This deals with the general areas of fission product chemistry in containment and hydrogen behaviour. Radioactive iodine can be the dominant fission product in terms of radiological consequences and thus its behaviour in postulated accidents is described in a conservative fashion. The Three Mile Island-2 accident demonstrated that the potential for radioiodine release in water reactor accidents is significantly lower than previously assumed. This prompted the acceleration of research programs in several countries.

The Canadian fission product chemistry program has the objectives of developing a fundamental understanding of radioiodine chemistry such that the gas-phase/liquid-phase partitioning of radioiodine can be quantified within an order of magnitude and of developing means of controlling the gas-phase partitioning. The program has provided a good understanding of iodine behaviour in terms of classical chemistry. In the last two years the emphasis has been placed on studying the behaviour of iodine in the presence of an intense radiation field. A major new facility to conduct semi-scale integrated tests is becoming operational at WNRE. Research with this facility complements that performed on a laboratory scale.

Hydrogen is produced in a severe accident if the Zircaloy fuel sheaths are raised in temperature to a point where rapid oxidation occurs by reaction with the surrounding steam. Oxidation strips the oxygen from the steam molecule, leaving hydrogen. Subsequent burning of the hydrogen in air can cause a pressure rise in containment and can damage components required for post-accident management. If the concentration of hydrogen in air exceeds certain limits, combustion can lead to detonation and elevated transient pressure.

A comprehensive program to investigate the behaviour of hydrogen in containment has been on-going at WNRE for a number of years. The program has generated the understanding necessary to develop and validate the VENT computer program used to verify containment integrity under accident conditions. Furthermore the program has demonstrated the effectiveness of the hydrogen mitigation system installed in CANDU reactors. Work is continuing on providing further data for code validation and to define the conditions which lead to flame acceleration and transition to detonation. There is general consensus, however, that hydrogen combustion is sufficiently well understood, and thus, less emphasis will be placed on this R&D area in the future.

D. Calandria Tube Integrity Program

This has the objective of quantifying the margin existing against failure of the calandria tube under the full range of potential failure modes for pressure tubes. As discussed in Section 9.1, if the calandria tube survives the rupture of a pressure tube the direct economic consequences are relatively minor. However a channel failure may lead to damage to in-core structures and thus be very significant from an economic viewpoint. Experiments are performed at full system pressure (10 MPa) using a full scale fuel channel. The tests show that following pressure tube rupture, the annulus space between the pressure tube and the calandria tube rapidly fills with the discharged two-phase mixture. The momentum and energy changes associated with this rapid process translates into straining of the calandria tube (waterhammer effect). The tests have shown that the annulus over-pressure depends upon the coolant subcooling; the more subcooled the water, the greater the transient over-pressure. However the accompanying calandria tube strain significantly alleviates the over-pressure. The experimental program and the associated model development effort are approaching completion, and have been instrumental in explaining calandria tube behaviour following rupture of pressure tubes in channel G-16 of Pickering NGS A Unit 2 and channel N6 of Bruce NGS A Unit 2.

E. Safety Thermal-Hydraulics Program

This provides experimental data needed to further develop and verify the computer codes used to simulate the system response to accidents. Most of the experiments are carried out in the RD-14 loop of the Cold Water Injection Test (CWIT) facility and the Large Scale Header Test Facility.

The RD-14 loop provides data for the verification of system codes while the other facilities generate data required for the verification of component models. As an example, the CWIT facility has provided much useful data on the refilling of feeders and fuel channels at conditions similar to those predicted for postulated large break LOCAs. The RD-14 facility is a large scale, full height representation of a typical CANDU heat transport loop and includes one fuel channel per coolant pass. Tests are performed at conditions representative of large and small loss of coolant accidents with and without forced coolant circulation. In the future some of these tests will be repeated in a modified RD-14 loop which includes five simulated fuel channels per coolant pass.

F. Moderator Circulation Test Program

This complements the moderator temperature measurements taken in the Bruce NGS A and Pickering NGS B reactors. The purpose of the program is to provide data suitable for further validation of moderator circulation codes (see Sect 4.2.3.2) particularly for conditions predicted to occur in postulated accidents. The test facility, now being constructed, is a model representation of a reactor calandria with one-quarter of the radius and a small thickness. This "slice" of a calandria will provide eventually two-dimensional distributions of fluid velocities, fluid temperatures, turbulence levels etc. for a variety of accident conditions that cannot be realistically reproduced in a power reactor. The program was initiated in 1987.

G. Fuel Channel Critical Power Program

This has two main objectives. The first is to perform experiments which provide data for the development and verification of CHF (Critical Heat Flux) and post-CHF correlations for use in accident analyses. The second is to perform experiments which provide data required to verify subchannel codes. The verified models are then used in the analyses of accidents such as loss of regulation and loss of coolant to determine the possibility and timing of the onset of degraded fuel cooling, e.g., are the thermal-hydraulic conditions such that the fuel is no longer well cooled by the essentially liquid coolant? The first objective has been largely achieved. A series of tests has been carried out with a test assembly having a heated length of 6 m and a flow geometry that is essentially identical to a fuel channel containing 37-element fuel. The CHF, post-CHF and pressure drop correlations for 37-element fuel were developed largely with data generated in this program. An extensive program is currently underway to achieve the second objective, i.e, to have a well verified subchannel code capable of predicting the onset of CHF, and post-CHF fuel temperatures, over a wide range of thermal hydraulic conditions.

4.3 Safety Assessments

4.3.1 Introduction

There are a number of low probability common mode incidents which, if protection were not provided against their effects, could lead to damage to station equipment, and possibly lead to radioactive releases above allowable limits. These postulated events are listed in Table 2-3.

The general philosophy is to limit the consequences of such events in order to maintain the essential capabilities to shut down the reactor, remove decay heat, limit radioactivity release and monitor the status of important plant parameters. Either Group 1 or Group 2 systems provide these capabilities.

The safety assessments include a review of the plant design. This provides assurance that the essential capabilities can in fact be maintained. Further, failure scenarios are identified which require detailed accident analysis. This provides the assurance that appropriate release limits can be met following any of the postulated common mode incidents.

In this section, a brief description of the safety assessments for each common mode incident considered in the design of Darlington NGS is provided.

4.3.2 On-Site Explosion

The worst hydrogen explosion in the tritium removal facility is postulated. Effects of overpressure and associated fires are assessed to demonstrate that Group 2 systems will survive. Radiological hazards as a result of tritium releases from the damaged tritium removal facility are evaluated to ensure that appropriate release limits are met.

4.3.3 Off-Site Explosion

Detonation of 61,500 kg TNT equivalent on the CN rail line at the closest approach to the station is postulated. Effects of overpressure, associated fire, and missiles on the station are considered. The review includes an assessment to determine the failure scenarios for radiological consequence calculations. It also confirms that all important structures and Group 2 systems will survive the peak overpressure.

4.3.4 Turbine Breakup

Missiles generated from turbine disintegration are postulated. Effects of missiles on safety related equipment are evaluated. A review of the layout and a probabilistic assessment of missiles striking critical equipment are performed to demonstrate that all Group 2 systems would survive and radioactive releases from damaged systems would be within allowable limits.

4.3.5 Internal Fires

Major combustible and ignition sources are identified and fires are postulated at various locations within the plant. Fire prevention strategy and fire suppression capabilities are reviewed. Effects of fires are assessed to demonstrate that systems in Group 1 or Group 2 or both would remain available, and no significant consequences to the public would result.

4.3.6 Design Basis Tornado

A tornado sweeping through the station is postulated. This is a low probability event. Impacts of high wind speed , rapid pressure changes and tornado generated missiles such as airborne cars and pipes are considered. Important station structures are confirmed to have been designed to withstand the wind and pressure effects. A review of the layout is carried out to ensure that Group 2 systems will remain available, and appropriate failure scenarios from missile damage are identified for radiological consequence analysis.

4.3.7 Design Basis Earthquake (DBE)

Strong ground motions from an earthquake are postulated to occur at the station. The assessment includes a review of the layout to show that the seismically qualified systems would survive, and would not be adversely affected by non-qualified system/component failures. It also identifies failure scenarios for detailed accident analysis.

A plant walk-through is normally carried out before the plant goes into operation to provide a final check of the appropriateness of equipment layout.

4.3.8 Airplane Crash

Probabilities of crashes of airplanes of various sizes are estimated. Credible crashes are postulated and consequences of airplane impact on structures and equipment are assessed. Large plane crashes on critical structures are shown to be incredible and consequences of credible small plane crashes are demonstrated to be acceptable.

4.3.9 Toxic/Corrosive Gas Rail Line Accident

Effects of toxic/corrosive gas released at the nearby rail line on the operating staff and equipment are assessed. It is shown that adequate means of detection is available and there is no significant hazard to the operators. Effects on equipment would develop slowly and all Group 2 system equipment would be unaffected.

4.4 Risk Assessment

4.4.1 The Evolution of Risk Assessment in Ontario Hydro

Risk assessment is an approach to safety management that derives from a formal expression of the concept of risk. Risk is defined as the product of event frequency and event consequence and is expressed in terms of units of consequence per unit of time. Inherent to this approach is the presumption that a frequent event of low consequence or an infrequent event of high consequence can be said to pose the same quantitative "risk" if the products of frequency and consequence are equal. It also involves recognition that risk can be reduced by engineering methods directed at reduction of event frequency, event consequence or both. Risk assessment is the process by which accident sequences are identified, their frequencies and consequences estimated and all risk contributions summed to obtain an integrated result.

The application of the concept of risk in reactor design can be found in the approach used by AECL in the early 1960's for the NPD and Douglas Point stations. Risk to society from other industrial activities was used as a basis to develop safety targets for the nuclear plant, from which design limits for initiating event frequencies and for major protective system unreliabilities were derived.

These ideas provided the basis for the development of AECB licensing guidelines, culminating in the publication of the Siting Guide (Reference 1). Although retaining elements of a risk approach, the setting of both frequency and consequence limits for two classes of event, "single" and "dual" failures, is essentially deterministic in nature.

A return to a more risk-based regulatory approach was considered in the late-1970's in the form of the Inter-Organizational Working Group guidelines (Reference 2). However, because of concern over public acceptability and doubts about the probabilistic techniques then available, the proposals were not implemented. Instead, in 1980, the AECB issued a consultative document C-6 (Reference 3), which retained the concept of fixed event classes but increased the number from two to five. In fact, C-6 had a reduced probabilistic content in that frequency limits for the event classes were not established. The AECB's Advisory Committee on Nuclear Safety have since issued a document, ACNS-4 (Reference 4), which recommends acceptance criteria based on event consequence and summed event frequencies.

To allow for a more thorough analysis of service system failures and to ensure that multiple failures did not pose an unacceptable hazard an additional, a probabilistically-based design review process was introduced for Bruce A, referred to as Safety Design Matrices (SDMs). The process involved the generating of event sequence and supporting fault tree analysis for a number of defined initial failures. Each sequence was followed until stable plant conditions were achieved, the consequences were shown to be acceptable, or the sequence frequency fell below 10^{-7} events/year. Failing this, design changes were implemented. These studies were expanded on Pickering B and Bruce B to include most major potential accident initiators. While SDMs do not constitute risk assessments in today's terms, they represented an important advance in introducing probabilistic methods into CANDU safety assessment.

Recent years have seen a rapid improvement in the quality of methods and data available for risk assessment. For Darlington, Ontario Hydro is presently completing the Darlington Probabilistic Safety Evaluation. This study is a comprehensive, integrated risk assessment whose primary purpose is to verify design adequacy, an objective achieved through the process of generating risk estimates for the station.

4.4.2 The Risk Assessment Process

The term risk assessment can be applied to any organized approach that has as its objective the quantification of risk. When used in the context of a nuclear power station, it is generally taken to imply a particular form of analytical process, illustrated in Figure 4-3.

The process can be separated into a number of stages for illustrative purposes, although in reality they are highly interdependent. These stages are:

(a) Definition of objectives, scope and identification of important initiating events;

(b) Development of event sequences, identification of mitigating system failure modes and development of system fault tree models;

(c) Identification and categorization of event sequence consequences and development of fault tree logic for each consequence category;

(d) Integration of system fault trees and consequence category logic;

(e) Evaluation of consequence for each consequence category and the combination of category frequency and consequence estimates to obtain integral risk estimates.

The objectives of a risk assessment have an important influence on the level of analysis performed, the magnitude of the task and the results obtained. For a nuclear reactor the consequence of concern may be core damage, release of radioactivity to the atmosphere, financial loss, public health effects, or all of these. The study may or may not choose to include the contributions from events with origins outside the plant such as earthquakes, other natural phenomena, aircraft crash, etc. Having established the scope it is necessary to identify, first in broad categories and then in more detail, all the various types of failures

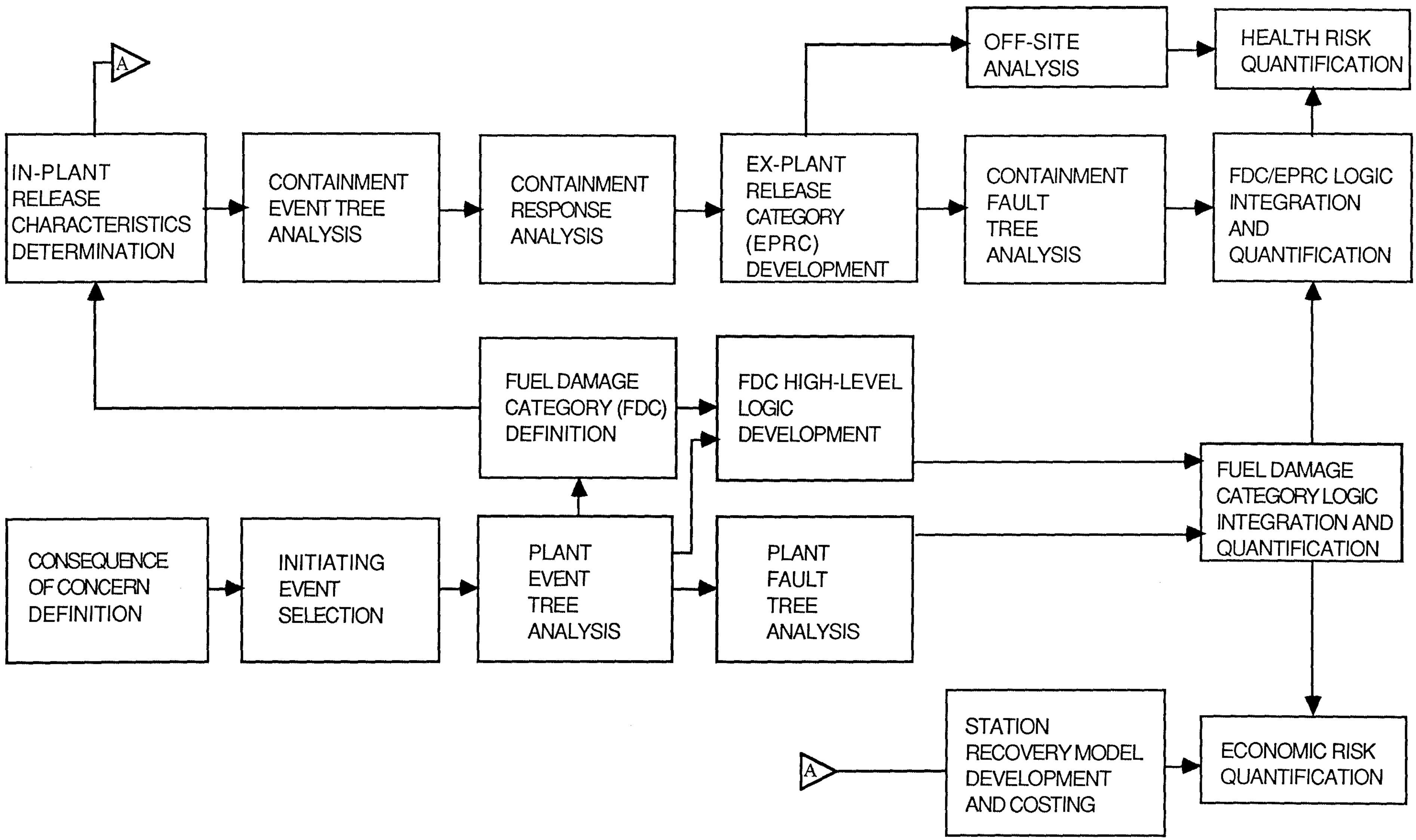

FIGURE 4-3 DPSE RISK ASSESSMENT PROCESS

that could initiate a sequence of events potentially leading to the undesired consequence. This is done partly by deductive logic, partly by review of system design and partly by observation of operating experience. These failures are known as "initiating events".

For each initiating event or group of events with a unique impact in terms of required mitigating system response, an event sequence diagram known as an "event tree" is produced. The purpose of the event tree is to identify the range of possible outcomes of the sequence, allowing for the possibility of failure of each or all of the available front-line mitigating systems. Each possible outcome of the event tree is allocated to one of several consequence categories depending on severity. The event tree is also used to define system failure modes for subsequent system fault tree analysis. A fault tree is a deductive logic diagram which reflects the structure and functions of a system and which defines the minimum conditions necessary to cause failure of the system. By assigning probability values to the components of the tree an estimation of the probability of the defined failure can be derived. Fault trees are generated for each of the front-line process systems, safety systems and support systems. The models are developed to a level of detail such that available component failure data may be applied. An important feature of these fault trees is the identification of the opportunities for human error and the quantification of human error probabilities.

As mentioned above, each event sequence outcome is assigned to one of several categories within which the consequences are judged to be similar. These groups of sequences are known as "consequence categories" which together cover the range of consequences as defined by the study scope. Categorization greatly simplifies the consequence modelling and fault tree integration tasks. Each consequence category is treated as a single event for consequence calculations and only those system failures which give rise to a change in consequence category need be distinguished. Fault tree logic is developed for each consequence category, down to a level where system fault tree models are available. If the risk assessment scope involves the need to include containment performance, two sets of consequence categories are required, one to categorize fuel damage and the other to categorize releases to the environment. The fuel damage categories serve as initiating events for containment and a containment event tree is produced, the outcomes of which form the basis for ex-plant release categories.

It is the process of linking plant models and accident consequences through consequence categories, and the solution of the fault tree logic to obtain frequency estimates for each category which distinguishes risk assessment from other types of reliability studies. The task is known as "integration" and involves the solution of combinations of fault trees for process, mitigating and common support systems, some of which can be very large. It is at this stage that any cross-links between systems caused by common components, services or human actions are accounted for. Similarly, the possibility that an initiating event may also impair a mitigating system is included, such as through the creation of a hostile plant environment by steam or liquid release.

The integration process generates frequency estimates for the consequence categories. Those event sequences which are the most important contributors to each category are identified, providing a basis for design adequacy review or assessment of emergency operating procedures. Each consequence category is defined in terms of representative characteristics of radiological releases, from which the consequences of concern to risk estimation can be evaluated. The summation of the product of frequency and consequence for each category leads to the final estimates of risk.

4.4.3 The Darlington Probabilistic Safety Evaluation (DPSE)

The study was initiated by Ontario Hydro with the following principal objectives:

(a) To provide a thorough safety design verification of the Darlington station;

(b) To identify those initiating events and accident sequences that dominate public safety and economic risk to the utility;

(c) To provide a comprehensive and realistic information base for the preparation of commissioning and operating procedures and for the training of personnel in handling accident situations;

(d) To provide those system reliability and event sequence assessments required as part of the licensing process.

The DPSE is concerned primarily with those failures which can pose a threat to the integrity of fuel in the reactor core. Events which originate outside of the station are excluded, except for loss of off-site power, on the grounds that design adequacy can better be demonstrated using deterministic methods. These studies were described in Section 4.3. The results of the DPSE study are presented in the form of health risks to the public and economic risk to the utility.

The DPSE has employed methods which represent, and in some ways advance, contemporary risk assessment technology, especially with respect to fault tree analysis and integration. System modelling has been exceptionally detailed and comprehensive. Special methods to identify and quantify human interactions have been developed. The consequence analysis covers a wide range of degree of fuel damage in order to assess the design from both a public safety and economic perspective. Economic consequence calculations aimed at identifying and quantifying station recovery tasks and resulting costs have been carried out. Radiation dose to the public both collectively and to the individual has been estimated. The DPSE is scheduled for completion in 1987.

4.4.4 DPSE Approach to Severe Accidents

The term "severe accident" is usually taken to mean an event which leads to physical damage to the core structure to a degree that fuel cooling cannot be assured. An example would be a total loss of heat sinks potentially leading to multiple fuel channel failures. Consequence analyses for this type of event can be shown to be unnecessary for licensing analysis, because the events normally involve multiple failures and fall outside the scope of the single/dual failure criterion. The implication is that such events are expected to be extremely unlikely. The ultimate consequence of a severe accident is independent of the number of mitigating systems available, because such systems can always be postulated to fail. The introduction of additional mitigation serves primarily to reduce the likelihood that the ultimate consequence will occur. Thus, the question of design adequacy revolves around the issue of whether the estimated frequency of a severe accident is sufficiently small.

In the DPSE approach, the view is taken that, provided the integrated frequency of severe accidents can be shown to be sufficiently small, no detailed assessment of consequence is necessary to determine design adequacy. The procedure being followed is to identify all the various sequences which potentially result in a severe accident and collect them within a single category using the integrated event tree/fault tree process previously described.

A program of work is currently underway within Ontario Hydro to develop risk-based corporate safety goals. A severe accident frequency target will be developed from the safety goals to establish design adequacy in this area.

4.5 Commissioning of Safety Related Systems

The last step in the design verification of the safety design is the commissioning of the safety related systems. Commissioning, in conjunction with the accident analysis, demonstrates that the design intent is met. To provide this demonstration the commissioning tests should be performed under conditions representative of those under which the systems are designed to operate. This is readily achievable for the process systems; it may not be practical, however, for the safety systems, which are designed to operate only under abnormal or accident conditions. The commissioning program for these systems is therefore designed with appropriate constraints related to public and operating staff safety, and economics.

The constraint of safety is satisfied by not performing tests which could result in unusual hazards for the operating staff or release of radioactivity. The economic constraint is satisfied by not performing tests which, by design, could require subsequent extensive inspection and, possibly, repair. Furthermore, the constraint of economics is satisfied by "type-testing" which avoids the need to perform separate commissioning tests on identical components and systems. This method is normally used to demonstrate some fundamental aspect of the design such as pump head and flow characteristics, thermosyphoning, etc.

Although, in general, attempts are made to verify the integrated function of the safety system, sometimes only testing of specific aspects of that safety function is practical. Testing of specific aspects is always complemented by simulations performed with the same computer codes and mathematical models used in accident analysis. The test conditions are selected to allow validation of these computer codes and permit reasonable extrapolation to the conditions of the accident scenario which forms the design basis for that safety system or function. A brief description of typical commissioning tests performed for the special safety systems follows.

The commissioning program for the Shutdown Systems include extensive wiring and functional tests of components. The reactivity worth of each shut-off rod is measured. An integrated test of both SDS-1 and SDS-2 is performed by tripping the reactor and measuring the consequent power rundown transient. These tests are normally performed with the system configured as assumed in accident analysis (some devices unavailable). The resulting power rundown transient is simulated with reactor physics codes.

The commissioning of the Emergency Coolant Injection system include tests intended to verify the operation of the initiation logic, the injection flow path, the operation of the rapid cooldown of the steam generators as well as tests to verify the reliability and maintainability of the system. An example of a "type-test" performed for the Pickering NGS B ECI system is the demonstration that important valves close within the time limit assumed in accident analysis. This test was performed at Ontario Hydro Research Division's pump test complex in a loop designed for 13.7 MPa, 315°C operation. An example of an integrated test is the "hot injection test" performed at Pickering NGS B. The test was conducted with the heat transport system (containing light rather than heavy water) pressurized at 8.7 MPa and 166°C, and no fuel in the reactor core. A signal, simulating the one that would be received in the event of a loss-of-coolant accident initiated the operation of the system. The injection of emergency coolant lasted for approximately 17 minutes.

The commissioning program for the Containment system typically includes vacuum building pressure tests and verification of the correct operation of the pressure monitoring and isolation system, the water spray system, the pressure relief valves, etc. A "type-test" of particular interest is the "blow through" test performed at Bruce NGS B. In this test the structural response of the vacuum building, the simultaneous operation of the pressure relief valves and the operation of the dousing system were demonstrated.

In conclusion, a comprehensive commissioning program for the safety systems is carried out in each Nuclear Generating Station for the purpose of demonstrating that the design intent is met. In particular these tests provide a demonstration that the assumptions used in accident analysis with respect to safety systems operation are valid.

REFERENCES

(1) D.G. Hurst and F.C. Boyd, "Reactor Licensing and Safety Requirements", Paper 72-CNA-102, Presented at the 12th CNA Conference, Ottawa, June 1972.

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(3) Atomic Energy Control Board, "Requirements for the Safety Analysis of CANDU Nuclear Power Plants", Consultative Document C-6, AECB Proposed Regulatory Guide, June 1980.

(4) ANCS, "Recommended General Safety Requirements for Nuclear Power Plants", Advisory Committee on Nuclear Safety, ACNS-4, June 1983.

Commissioning and Operations Management

5.0 COMMISSIONING AND OPERATIONS MANAGEMENT

5.1 Introduction

The overall objective of the Nuclear Generation Division is to produce electricity at the lowest possible cost for Ontario Hydro customers consistent with the achievement of acceptable standards of:

Worker Safety
Public Safety
Environmental Protection
Product Quality

Public safety refers to protection of members of the public against events which could result in injury, disability, or death caused by Hydro facilities or employees. Two basic classifications of acute events are maintained for Ontario Hydro facilities:

Conventional Events - where a member of the public is killed or injured as a result of conventional hazards, and

Radioactivity Events - where a member of the public is killed or injured as a result of nuclear accidents involving radioactivity.

This section describes the management of commissioning and operational nuclear safety.

5.1.1 Objective of Commissioning Nuclear Safety Management

Commissioning encompasses all those activities designed to demonstrate that equipment and systems perform within the intent of their design specification at the time they are declared in-service. It is Ontario Hydro's objective to assure that the quality of engineering, operation, maintenance, and other activities in the commissioning of nuclear power stations is consistent with the specified levels of safety and reliability. This is achieved by ensuring:

- equipment requirements are specified;
- adherence to applicable codes, standards and regulations;
- commissioning procedures to adequately check the system are prepared;
- training and qualification of all staff is appropriate;
- approved procedures, qualified staff, and supervisors are used in all phases.

- verification of procedures and field activities;
- all results, deficiencies and corrective actions are reported;
- maintenance of adequate records;
- procedures to be used in the operation phase are validated;
- a systematic review to provide commissioning completion assurance is undertaken, and
- activities are independently confirmed by audit.

The management process control elements used in the commissioning phase are identical to those used in the operational phase. These elements are described in Sections 5.2 through 5.9. Special mention should be made, however, of the Commissioning Completion Assurance phase. It is the responsibility of the Technical (or Commissioning) Manager to ensure that all commissioning procedures were adequate, appropriate and that all acceptance criteria were met and all deficiencies suitably resolved. When this is complete, the system or systems can be placed into operation.

5.1.2 Objective of Operational Nuclear Safety Management

Nuclear facilities are operated with the objective that the public risk resulting from potentially acute releases of radioactivity is maintained at an acceptably low level.

Work at a nuclear facility involves four general types of activities:

(a) operation;

(b) maintenance;

(c) modifications;

(d) surveillance.

These activities all have a potential impact on public safety if performed in an appropriate manner.

5.2 Management Elements

The following key elements are necessary to accomplish the objective of nuclear safety management:

(a) a defined operating envelope;

(b) defined responsibilities and limits of authority;

(c) work activities are performed by qualified staff to approved procedures;

(d) approved equipment and materials are used;

(e) work activities are documented;

(f) surveillance of work activities and equipment performance is undertaken and corrections are made as required;

(g) emergency preparedness is in place.

These elements are discussed in more detail below.

5.3 Operating Envelope

Work activities at nuclear facilities are executed within a defined operating envelope which is approved by the AECB. The plant is maintained in a condition that has been analysed and shown to be safe at all times. The Operating Policies and Principles (OP&Ps) constitute this operating envelope.

The OP&Ps:

(a) Identify the framework within which all activities must be performed;

(b) Identify the level of approval required for these activities.

5.3.1 Approved Procedures

The following is a description of two key procedures which ensure that operations are performed within the OP&P framework.

5.3.1.1 Change Control

Change control is the process whereby all planned deviations in plant operation or design (equipment, systems, computer software and procedures) are analyzed and approved. The objective of the process is to ensure that public safety is maintained.

The station's technical (or commissioning) section is responsible for administering the change control procedure. This involves preparing the documentation describing the change and ensuring that it is circulated for review by qualified personnel. The approval process cannot be circumvented just because the change is minor or temporary in nature. Permanent changes to safety related systems require involvement of the design organization prior to implementation. Similarly, changes to procedures are subject to the same review and approval process as the original document.

All permanent or temporary changes which could significantly affect public safety are reviewed and approved by the Station Manager. Prior approval by the AECB is also required for changes which affect the assessment of public safety.

The Shift Supervisor ensures that no changes are made without proper approval.

5.3.1.2 Work Control

Work activities are controlled at nuclear facilities by the work authorization process. It confirms that:

(a) the system/unit/plant is in a safe state for the work to be done from the point of view of nuclear safety and worker safety;

(b) required approvals are obtained for the work;

(c) equipment is tested for operational capability after any work;

(d) the Unit First Operator and Shift Supervisor are aware of significant work activities.

5.3.2 Deviations from Approved Procedures

Deviations from approved procedures are authorized by the shift supervisor unless higher authorization is required by the operating license or OP&Ps.

Most anticipate abnormal situations are covered by the Abnormal Incident Manual which provides the operator with procedures to return the unit to a safe state. However, an emergency situation may arise where corrective action must be taken immediately. There may be insufficient time for supervisory consultation without a significant risk to worker safety, public safety, environmental protection or major damage to equipment or facilities. In these circumstances, any deviation from approved procedures would be with the intent to get the unit back into a state where approved procedures could be used. The appropriate approval authority is then informed of the action taken.

5.4 Responsibilities and Limits of Authority

Personnel involved in the operation of nuclear facilities are trained to understand their responsibilities towards achieving the nuclear safety objectives and also clearly understand the limits of their authority.

5.4.1 Atomic Energy Control Board (AECB)

The Atomic Energy Control Board's (AECB) role as that of a regulator is to set the standards of acceptable public safety for the operation of nuclear facilities. Upon demonstration that the licencee has met these acceptable standards, an operating license is issued.

In essence, the requirements of the operating license are:

(a) key staff who operate the station must be approved by the AECB;

(b) station work activities are executed within OP&Ps;

(c) AECB approval must be obtained for any modification which may significantly and adversely affect public safety or otherwise changes the basis upon which the operating license was issued;

(d) the AECB must be informed promptly of events that significantly and adversely affect public safety.

The AECB does not design or operate nuclear facilities; they monitor and audit activities to ensure that the licensing requirements are met.

5.4.2 Station Manager

The person approved by the AECB in the Station Manager position has overall responsibility for plant safety.

The Station Manager ensures that the station is operated such that:

(a) staff are trained to execute their duties. Particular emphasis is placed on the authorized staff;

(b) the AECB is aware of any modifications to equipment or procedures, or equipment malfunction, which may have a significant adverse effect on public safety;

(c) the required level of analysis is done to justify changes in operation or design;

(d) the structure of the organization is designed to carry out work activities safely;

(e) approved procedures exist to control work activities.

The combination of the above is effective in maintaining a high quality of plant operational and commissioning safety.

The Station Manager's authority in safety and safety related areas is governed by the operating license.

5.4.3 Shift Supervisor

As senior (AECB) authorized person on shift, the Shift Supervisor is responsible for ensuring that the plant is in a condition that has been shown to be safe. This is achieved by ensuring that activities are performed within the OP&Ps and, in particular, by controlling change.

The Shift Supervisor ensures work is performed to approved procedures and must personally authorize activities that affect controlling of power, cooling of fuel and containing radioactivity. The Shift Supervisor reports through the production line of command to the Station Manager.

The authority of the shift supervisor is governed and defined by the OP&Ps and the operating license. The approval of the Station Manager must be sought for certain activities.

5.4.4 Shift Operating Supervisor

The Shift Operating Supervisor (SOS) position is an AECB authorized position. The SOS has the same reactor operation and safety training as a Unit First Operator, has additional administrative responsibilities over the Unit First Operator, but no additional nuclear safety authority. The SOS is responsible for supervising the operator work group to ensure that they function effectively within their authority limits.

5.4.5 Unit First Operator

There is an AECB authorized Unit First Operator for each operating unit. The Unit First Operator has responsibility for activities that occur on the unit and reports to the Shift Supervisor either directly as required by certain procedures or through the SOS.

The authority of the Unit First Operator is governed by the OP&Ps and limited by administrative controls. The approval of the Shift Supervisor is sought as required for certain activities.

5.4.6 Station Technical Support Group

A station technical group supports the day-to-day operation of the station. The technical support group includes system engineers who have specific plant systems assigned to them. The responsibilities of the technical support group are to:

(a) monitor, evaluate and report station operating data and events and prepare maintenance programs to ensure that systems achieve performance targets;

(b) troubleshoot operating problems and prepare solutions;

(c) prepare and obtain approval of nonstandard work procedures to be performed by production field staff;

(d) prepare change documentation for the Station Manager's approval;

(e) provide on-site technical support to the production staff;

(f) prepare, revise and obtain approvals for operating procedures, training documents and standard maintenance procedures;

(g) provide necessary interfacing with other organizations involved in managing nuclear safety (eg, with the centralized design and technical support departments).

Plans for performing work are reviewed by other affected work groups and system engineers to ensure overall nuclear safety and worker safety.

The technical support group has no direct authority or control of production activities.

5.4.7 Corporate Technical Support Groups

Strong, centralized, nuclear technical service and design departments at head office provide specialist and analytical support to the station technical group.

These departments interact as required to provide quality service to station staff. As with the station technical support group, the centralized departments have no direct authority or control of production activities.

5.5 Qualified Staff

Qualified and reliable staff by virtue of selection, training, testing, requalification and experience perform work activities at nuclear facilities.

A centralized staffing group is responsible for ensuring the right people will be available to the nuclear operating departments with the appropriate level of training and experience at the time they are required. This is accomplished by a recruitment program and by an in-house training program, conducted in dedicated training centres and supplemented by on-the-job training at each plant.

New hires into the nuclear program participate in fundamental science and engineering courses along with appropriate skills training, as required by job specifications for each employee classification. Additional refresher and upgrading training in these areas is provided as required.

Authorized staff undergo extensive training, including a rigorous selection and observation period by Ontario Hydro, prior to final approval by management and AECB authorization. The training encompasses general and specific examinations and training on full scope simulators and/or through actual plant operations. Each multi-unit nuclear station is equipped with a simulator which is modelled on the specific nuclear generating unit.

Training simulators provide simulated on-power operating experience to the operators for a wide range of conditions from normal operation, startups and shutdowns, through upset and severe accident conditions. Authorized staff complete simulator refresher courses semi-annually.

Additionally, operating staff are instructed using actual events described in Significant Event Reports (SERs) as training material.

5.6 Approved Equipment and Materials

Only approved equipment and materials are used for work activities. Materials used to repair, replace and modify are according to approved specifications for the component, equipment or system and according to relevant acts, codes and standards. Non-exact replacement components are approved by the design organization and the Regulatory Authorities where appropriate. History dockets are maintained for nuclear systems.

5.7 Documentation

Work activities at a nuclear facility must be documented. Adequate and retrievable documentation is required to demonstrate compliance with regulatory requirements, to facilitate monitoring and auditing of performance and to facilitate information gathering for the review of undesirable operating events. As a minimum, work activities are documented in the station log and/or on work reports. Commissioning Reports, Test Reports and In-Service Reports are examples of other available documentation.

5.7.1 Significant Event Reports

For any unplanned event that causes, or has the potential to cause, an undesirable impact upon station performance in the areas of Worker Safety, Public Safety, Environmental Protection, Product Quality or Product Cost, a Significant Event Report (SER) is prepared. The SER is written promptly by station staff and includes initial actions and observation, immediate investigation results, and the author's recommendations for further follow-up actions and manager's comments and direction. The SER also provides the mechanism to initiate detailed, specific investigation as required.

5.8 Surveillance

5.8.1 Station Surveillance

Periodic testing and inspection is performed to verify satisfactory performance of equipment and systems during normal operation.

Supervisors verify the work of subordinates. Direct surveillance of each and every activity is not required when qualified personnel are using approved procedures to perform the activity. However, periodic surveillance is carried out at an appropriate frequency to provide each supervisor with adequate confidence in the quality of the individual's performance.

Maintenance on safety systems devices is verified through testing, where possible, before returning the system to its normal state. Additionally, testing and inspection is used to substantiate that safety systems meet performance targets.

For other systems maintenance is verified by testing and/or inspection. In these cases testing will establish that the equipment is restored to specification, and that it is operating properly and to system requirements prior to declaring the equipment in service. Where possible testing includes a functional test of the system.

The station technical support group monitors station documentation regularly to identify performance trends of safety-related systems and process systems.

Significant Event Reports (SERs) are reviewed by station management and the technical support group to ensure appropriate actions are taken in a timely manner so that the probability of recurrence of the event is minimized. Follow-up actions arising from the events are identified in the SER documentation system and tracked by the technical support group.

Surveillance of change control is done by audits of the process.

A station quality assurance group monitors for conformance with prescribed standards and established management control mechanisms.

Station Health Physics and Radiation Control groups monitor and audit station performance in radiation protection practices.

Stations have an established management by objectives program in place to monitor and assess performance in five key areas, one of which is public safety. Objectives have been defined which are considered to represent good practice for the nuclear industry. Indices are selected which can provide a measure of performance. For each index a numerical standard has been established. Senior management use the program to detect poor performance relative to targets.

Annual station Quality Assurance program reviews are carried out involving key station management and corporate support management.

5.8.2 Nuclear Integrity Review Committee (NIRC)

The composition and role of the Nuclear Integrity Review Committee have been discussed in Section 1.

Significant events of serious consequence (actual or potential) are brought to the attention of the NIRC. These SERs are typically only about two to three percent of the total number written (ie, 676 written in 1986). The NIRC reviews the timeliness and completeness of corrective actions and also provides an overview of specific and generic issues. Additional corrective actions, or higher priority actions, may be assigned as a result of the Committee's deliberations.

5.8.3 Corporate Support Groups

Significant Event Reports and other data are analyzed by Corporate support groups for trends, or lessons, and the results transmitted to other affected groups in the utility's design, operations, and training organizations. This analysis includes trends in system faults, component failures, and human performance and results in continued improvement of procedures, training and equipment.

5.8.4 Human Performance Evaluation

The performance evaluation process is reviewed and upgraded as areas of potential improvement are identified. A recent example of such an improvement is in the field of human performance evaluation. In recognition of the fact that human errors make a significant contribution to the total number of significant events, a trial of the Institute of Nuclear Power Operations' (INPO) Human Performance Evaluation System (HPES) was conducted at two nuclear generating stations in 1985. The HPES is a system which enables the root cause and contributing factors of human performance related events to be determined in order that effective corrective actions can be developed.

The trial results indicated that the follow-up to significant events was generally adequate, although more detailed evaluation was useful for complex events or events that occurred over a number of shifts. As a result, a number of improvements in human performance related significant event reporting and evaluation are currently being implemented as follows:

(a) a training module in simple human performance evaluation techniques has been developed and delivered to shift supervisors;

(b) a worksheet has been prepared for use during evaluations. The worksheet provides a systematic and comprehensive approach to the determination of contributing factors;

(c) each nuclear generating station has sent a representative to INPO's Human Performance Evaluation System training course. The individual acts as a station resource in undertaking detailed human performance evaluations;

(d) a divisional database of human performance related events is being established. The database will enhance human error trending capability and allow emerging problem areas to be identified at any early stage.

Audits by Corporate groups are carried out to confirm conformance to the quality assurance program.

5.9 Emergency Preparedness

A managed system of equipment, procedures and qualified staff is in place to assure an appropriate response following a radiation incident at a nuclear facility so that adverse public safety consequences are minimized.

It consists of a trained station organization, approved plans and implementing procedures and the necessary facilities and equipment to minimize off-site consequences. This emergency response capability (discussed in Section 13) provides an integrated approach among station personnel, Corporate staff, local officials and the Province to quickly make and implement protective action decisions. This capability includes Provincial and Municipal plans and procedures and a training program for external participants. The response capability is routinely exercised and reviewed to identify possible areas for improvement.

Operations Training

6.0 OPERATIONS TRAINING

6.1 Introduction

Competent staff have played a key role in achieving excellent results in all areas of commissioning, operation, and maintenance of Ontario Hydro's CANDU units. Success in these areas depends on the integrated output of all employees in stations as well as technical support departments.

Since the beginning of the nuclear program, Ontario Hydro's strategy has been to train all its nuclear operations staff in common areas of system knowledge, science fundamentals, equipment principles, radiation protection and conventional safety. This common training forms the foundation for the more specific plant system training delivered at each department.

Comprehensive training programs exist for all categories of staff with particular emphasis on the four major families which comprise approximately 70% of the staff. The four major families are:

- Management and Professional (M&P) staff
- Nuclear Operators
- Mechanical Maintainers
- Control Technicians.

The training programs are categorized as follows:

Initial Training

The initial training program is delivered to all new hires in the above categories and has two major goals:

- To provide the trainee with the minimum knowledge and skills to safely perform initial work assignments and continued training at the work location to which he/she is initially allocated.
- To confirm that the trainee has the personal attributes and ability to perform initial assignments in an acceptable manner and to progress to the higher level positions in the category into which he/she has been hired.

Progression Training

Training continues after initial allocation to a work location to complete the requirements for progression to the first regular position within the classification. This consists of both common and department specific training.

Promotion Training

Any training beyond the first regular position to fill higher positions is referred to as promotion training. In most cases this training is department specific but may include common training such as supervisory.

Refresher Training

Refresher training includes periodic retraining or practice to maintain proficiency and can range from maintenance of welding skills, review of radiation protection procedures to regular practice by unit first operators on a full scope simulator.

Upgrading Training

Upgrading training can arise as a result of design modifications, improved procedures, new regulatory requirements, new tools and techniques, or where performance reviews indicate a need for new or improved training.

6.2 Training Organization

6.2.1 The Nuclear Staffing Group

The Nuclear Staffing Group is totally dedicated to the functions of staff planning, hiring, training development, and training delivery for Ontario Hydro's nuclear operations. It is made up of four Departments in three geographic locations. These are:

(a) Central Nuclear Training Department (CNTD)

Located at Head Office, Toronto, its main functions are to provide staff planning and hiring services for the nuclear operations departments and training development services for the nuclear training centres and department training sections.

(b) Western Nuclear Training Centre (WNTC)

The Western Nuclear Training Centre, located at the Bruce Nuclear Power Development, provides initial, progression, promotion, refresher and upgrading training to employees at the Bruce Nuclear Power Development. The training facilities include classrooms, skills shops, and the Bruce NGS A & B training simulators.

(c) Eastern Nuclear Training Centre

The Eastern Nuclear Training Centre, located at Pickering, provides initial, progression, promotion, refresher, and upgrading training services to employees located at Pickering NGS, Darlington NGS, NPD NGS and Head Office nuclear operations departments. The training facilities include classrooms, skills shops, and the Pickering NGS A & B full scope simulators. The Darlington NGS simulator will be installed at ENTC in 1988.

(d) Simulator Services Department (SSD)

The Simulator Services Department, located at Head Office, is responsible for commissioning and maintenance support of training simulators and computerized training aids to meet the needs of the nuclear training centres and department training sections.

6.2.2 Department Training Sections

In addition to the Nuclear Staffing Group (NSG), each Operating Department has a small training unit or section as part of its line organization responsible to the Department Manager for integration of the Department specific Training Program with the NSG Programs. This includes:

- Planning, scheduling, monitoring, and follow-up of all training to be done by the Department staff including on-the-job training.

- Preparation and delivery of department specific training using subject matter experts from production and technical sections and training development expertise from NSG.

6.3 Training of Major Job Families in Nuclear Operations

The following is a brief summary of the initial and progression training programs for each major family.

Management and Professional Staff

There are approximately 1300 Management and Professional staff in Ontario Hydro nuclear operations. Annual hiring into this job family occurs at the rate of approximately 50 people per year, largely to make up losses due to attrition. New hires are university graduates with degrees in engineering or honours science.

They receive initial training of 4 months duration at ENTC or WNTC. Following this, they spend 20 months at a field department (eg, nuclear station) receiving field training and development work experience.

After two years from time of hire and upon successful completion of the training program these staff assume working positions as Assistant Technical Supervisors.

From that point on, they receive on-the-job technical training and some supervisory training as required. They may be selected into the Shift Supervisor in Training program at a later date (see Section 6.4.3).

6.3.2 Nuclear Operators

There are approximately 1000 nuclear operators in Ontario Hydro. Annual hiring occurs at the rate of 30 per year, largely to make up losses due to attrition. Minimum education requirements for entry into this job family are Grade 13 with an emphasis on maths and science.

Nuclear oprators receive initial training of 4 months duration at ENTC or WNTC followed by assignment to a nuclear generating station for 20 months of field training and developmental work experience.

After two years from time of hire and upon successful completion of training program, these staff assume working positions as Assistant Nuclear Operators*.

From that time on, operators may be selected for further specialized training (eg, fuelling machines) or for promotion training to become SECOND Operators**.

Experienced operators trained to second operator qualification may be selected into the Unit First Operator in Training program at a later date, which is described in Section 6.4.1.

6.3.3 Control Technicians

These are composite technicians*** who can be trained and deployed to work on any of the control and electrical systems in Ontario Hydro nuclear facilities. There are approximately 800 Control Technicians in nuclear operations. Annual hiring into this job family averages 36 per year required to make up for losses due to attrition and for growth in staff level requirements.

* The primary role of an assistant operator is to execute any operating activities required in the field according to approved procedures under the supervision of a first or a second operator.

** The primary role of a second operator is to execute or supervise the execution of all operations in the field according to approved procedures as required by the First Operator to operate the unit.

*** Ontario Hydro utilizes composite trades for both its Control Technician and Mechanical Maintenance job families. This policy was instituted to reduce individual radiation dose and allow flexibility in the assignment of staff to various jobs.

New hires must have completed Grade 12 and had further concentrated study of electronics, electricity or industrial process instrumentation at a level consistent with a 2 or 3 year community college course or have a completed apprenticeship (or equivalent) in industrial process instrumentation or electrical power and control equipment maintenance. They receive initial training of 9 months duration at ENTC and WNTC, combining classroom studies and hands-on shop assignments.

Following successful completion of the NTC training, technicians are assigned to a nuclear facility where alternating training and work assignments are given. They progress through training and work experience to become qualified Shift Control Technicians after approximately 4 years from time of hire.

They continue to progress and may, in the longer term and as vacancies arise, be selected to become Senior Shift Control Technicians and receive further training and supervisory skills.

6.3.4 Mechanical Maintenance

These are composite tradesmen who can be trained and deployed to work on any of the mechanical equipment and systems in Ontario Hydro nuclear facilities. There are approximately 800 Mechanical Maintenance staff in Ontario Hydro nuclear operations. Annual hiring requirements are approximately 25, largely to make up for losses due to attrition.

New hires must have completed Grade 12, preferably in the four year mechanical technology program, with emphasis on mathematics, science, and shops. They may be hired on at an advanced step if they have a completed apprenticeship in such areas as machine shop practice, pipefitting, welding, etc, with two or more years of pertinent industrial experience.

They receive initial training of 9 months duration at ENTC or WNTC consisting of classroom lectures and workshop training.

They are next assigned to a nuclear facility where alternating training and work assignments are given. They progress in this way to become Journeyman Mechanical Maintainers after approximately 6 years from time of hire.

As they accumulate work experience and as vacancies arise, they may be selected to become working supervisors and will take the required supervisory training.

6.3.5 Safety Training

Members of the major job families receive 186 hours of safety training. This includes extensive training in radiation protection as well as training in conventional industrial safety.

In addition to the above, selected volunteers are given comprehensive training in firefighting and first aid.

6.4 Training of Nuclear Station Control Room Staff

The structure of the operating staff is shown in Figure 6-1. A Shift Supervisor is responsible for the day-to-day operation and maintenance of the nuclear station. The Shift Supervisor is normally a graduate engineer but may also be a previously authorized Unit First Operator who received additional technical and station specific training.

The Shift Operating Supervisor is responsible for the operating aspects of the plant and normally has six First Operators reporting to him. Each unit (normally 4 in a multi unit plant) is operated by a Unit First Operator. In addition, one First Operator oversees the operation of common systems and another the operation of fuelling machines.

Staff assigned to Shift Supervisor, Shift Operating Supervisor and First Operator positions must be authorized by the Station Manager and approved by the Atomic Energy Control Board.

6.4.1 Unit First Operator Training Program

Experienced operators trained to Second Operator qualification may be selected to the Unit First Operator Training Program. These assignments are given to the best performers.

The Unit First Operator training program is outlined in Figure 6-2. Completion of this program includes 5 written examinations set by the Atomic Energy Control Board. When a candidate successfully completes the training program, he/she is authorized by the Station Manager with the approval of the Atomic Energy Control Board. The authorization is specific to the station where it was obtained.

General Training

The general training consists of two distinct programs dealing with nuclear and conventional topics separately. These programs are common for all nuclear stations in Canada and therefore are not repeated when a first operator is assigned from one station to another.

The programs consist of approximately 75 percent classroom training and approximately 25% self study. At the end, after meeting the Ontario Hydro requirements, the candidates must successfully complete an AECB audit exam before proceeding to the next phase of the training.

The content deals with the science fundamentals and system and equipment principles applicable to Canadian nuclear stations.

Station System Standard Operation

The standard operation training is based on the station operating documents dealing with operating policies and principles, station system startup, operation and shutdown, system testing and chemistry control.

This 13-week program gives each candidate sufficient unit knowledge and operating skills to be able to operate the unit on a day-to-day basis under normal conditions. A full scope simulator which replicates the control room of the station is used to develop the skills and assess the performance of the candidates during this program since unit startups and shutdowns are infrequent in the station. Ontario Hydro has a full scope simulator for each of its multi-unit stations.

Integrated Systems Operation

This portion of the program deals with non standard operation of individual systems, overall unit upsets, loss of auxiliary services and emergency operation unit accident conditions and is based on the station operating documents. This 16-week simulator-based program gives each candidate sufficient knowledge and skills to diagnose and respond to minor or major station malfunctions or upsets without reference to documentation for the first few minutes. After the initial actions have been taken, candidates should then identify and refer to appropriate documentation.

The training program covers all events in the Abnormal Incidents Manual, with approximately 70% covered by simulator exercises. The remaining events are covered in the classroom.

During this program, once candidates satisfy the Ontario Hydro requirements, they must successfully complete two Atomic Energy Control Board audit exams (Nuclear and Conventional Plant Specific).

Radiation Specific

This portion of the program deals with station radiation effluent monitoring, radiation monitoring within the plant, emergency radiation procedures and radioactive work control based on the station procedures. This 8 week program gives the candidate the knowledge and skills to be able to effectively oversee and control radioactive work on the unit, monitor effluent releases and act in case on an emergency.

Once Ontario Hydro requirements are satisfied, the candidate must successfully complete the last AECB audit exam before proceeding to the co-piloting phase.

Throughout the program candidates periodically return to their crews where they receive additional assignments related to the Unit First Operator job. In addition a number of field checkouts must be completed which gives crew supervision an opportunity to assess the candidates' performance.

Co-Piloting Phase

This last phase of the program is an opportunity for the candidate to perform all the duties of a Unit First Operator under the observation of an existing qualified operator. It also gives the shift supervision and station management an opportunity to assess the candidate on a periodic basis during this 3-month period. Upon successful completion of this phase, the Station Manager will formally request approval from the Atomic Energy Control Board to authorize the candidate.

6.4.2 Shift Operating Supervisor Training Program

All candidates for this position are previously authorized Unit First Operators. The additional training requirements are mainly administrative and supervisory in nature and the program duration is 4 months.

6.4.3 Shift Supervisor Training Program

Candidates are selected from the best performers in the station technical section. In a few cases selections are made from exceptional operator or maintenance staff. The basic layout of the Shift Supervisor program is the same as the Unit First Operator. The difference lies in the depth of knowledge expected in each area and the role the Shift Supervisor assumes during the simulator portion of the program and during the co-piloting phase.

6.4.4 Refresher Training Program

The base load nature of Ontario Hydro nuclear generating stations, together with on-power fuelling, contribute to their high capacity factors. Consequently the control room staff have relatively little exposure to such procedures as plant startup and shutdown, not to mention emergency procedures.

To maintain the desired level of proficiency in control room operations, the Shift Supervisor, Shift Operating Supervisor, and Unit First Operators of each crew receive periodic refresher training. Currently the training is two weeks per year. This includes unit startup and shutdown, plant transients and emergency procedures. Refresher training is conducted on a full scope replica simulator where the crew members, as a team, must carry out specific exercises. During the training, in addition to reviewing operating procedures, there is repeated emphasis on the science fundamentals and operating principles. The value of full scope simulator refresher training was demonstrated on August 1, 1983 when Unit 2 at Pickering station was shut down normally without the need for safety systems following a pressure tube rupture. This type of incident had been practiced on the simulator as part of refresher training.

To ensure emergency preparedness of the remainder of station staff refresher training is also conducted in the field to ensure that field operators, maintenance and technical staff and control room staff can interact and work as a team in the event of a plant transient or emergency. This training ranges from radiation emergency, fire and conventional accident drills to response to a simulated unit transient. Refresher training is also conducted in specific areas (ie, welding, work protection) when needs arise.

6.5 Training Design

A fundamental component of Ontario Hydro's nuclear training program is the systematic design and development of training. While Ontario Hydro's nuclear training programs have always incorporated both knowledge and performance requirements, it was recognized in the late 1970's that a more rigorous approach to training design was required. This approach utilizes the following six steps:

1. Analyze the Job
2. Determine the Training Objectives
3. Develop Appropriate Training
4. Conduct the Training as Designed
5. Test to Ensure Objectives are Met
6. Evaluate the Effectiveness of Training

Utilizing this approach increases our confidence that the training programs are performance based and are effective and efficient.

Major projects are currently in progress to re-evaluate the design of the Unit First Operator and Nuclear Shift Supervisor Training Programs.

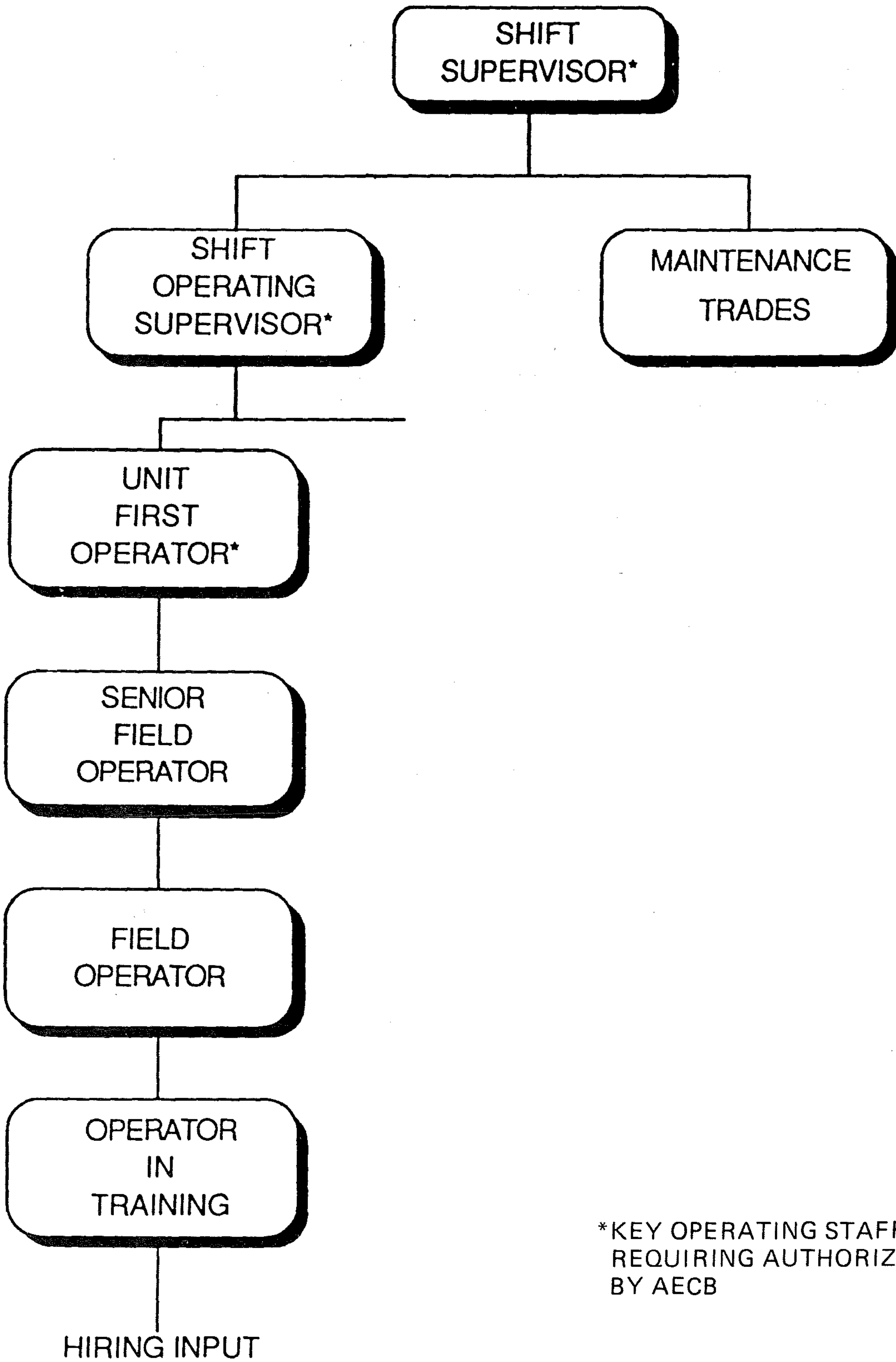
6.6 Summary

Effective training is not the only contributor to good operational results, but it is an essential one.

The results of Nuclear Operations to date suggest that past training programs have been effective. However effective training today and in the future cannot be assumed on the basis of past results.

Training programs must be updated to reflect operational experience and changes in technology. Ontario Hydro will continue to develop, revise, and improve the training programs for Nuclear Operations in order to ensure that they remain effective.

OPERATING STAFF STRUCTURE



*KEY OPERATING STAFF REQUIRING AUTHORIZATION BY AECB

FIGURE 6-1

UNIT FIRST OPERATOR TRAINING PROGRAM

SELECTION — 12 MONTHS

NUCLEAR GENERAL (12 WEEKS)		CONVENTIONAL GENERAL (13 WEEKS)		ON SHIFT(27 WEEKS)
-SCIENCE FUNDAMENTALS -EQUIPMENT AND SYSTEMS -75% CLASSROOM -25% SELF STUDY -WRITTEN CHECKOUTS	AECB EXAM	-SCIENCE FUNDAMENTALS -EQUIPMENT AND SYSTEMS -75% CLASSROOM -25% SELF STUDY -WRITTEN CHECKOUTS	AECB EXAM	-WORK ASSIGNMENTS

STATION SYSTEM (13 WEEKS) STANDARD OPERATION	ON SHIFT (6 WEEKS)	STATION SYSTEM(16 WEEKS) INTEGRATED PLANT OPERATION	
-SYSTEM TRAINING -STARTUP/SHUTDOWN -TESTING -50% CLASSROOM -25% SELF STUDY -WRITTEN/SIMULATOR CHECKOUTS	-FIELD CHECKOUTS	-SYSTEM MALFUNCTIONS -UNIT UPSETS -EMERGENCY PLANT PROCEDURES -40% CLASSROOM -30% SIMULATOR -30% SELF STUDY -WRITTEN/SIMULATOR CHECKOUTS	AECB EXAMS (NUCLEAR AND CONV)

36 MONTHS

ON SHIFT (30 WEEKS)	RADIATION PROGRAM (8 WEEKS)	ON SHIFT(17 WEEKS)	CO-PILOT (12 WEEKS)
-WORK ASSIGNMENT	-SEMINARS -FIELD SKILLS -EMERGENCY PROCEDURES -CLASSROOM TRAINING -WRITTEN/FIELD CHECKOUTS	-WORK ASSIGNMENTS	
	AECB EXAM		AUTHORIZATION

FIGURE 6-2

Employee Radiological Health and Safety

7.0 EMPLOYEE RADIOLOGICAL HEALTH AND SAFETY

7.1 Introduction

Ontario Hydro is committed to achieving acceptable standards of health and safety for its employees by effectively managing all risks resulting from or associated with its activities and operations (Reference 1). A set of radiation protection policies and principles (Reference 2) have recently been published to provide the basis in the future for the Ontario Hydro radiation protection program.

The AECBs regulations (Reference 3) and Ontario Hydro's radiation protection program are based on the recommendation of the International Commission on Radiological Protection (ICRP), an international agency, comprised of internationally recognized experts in radiation protection. This group has been recognized as the appropriate primary source of guidance on radiation protection since 1928. In particular, Ontario Hydro subscribes to the principle of maintaining radiation exposures as low as reasonably achievable (ALARA).

This section briefly describes that part of Ontario Hydro's radiation protection program for protection of employees and provides information on the results which the program has achieved in providing a high standard of radiological health and safety for employees at nuclear facilities.

7.2 Radiation Protection Program Objectives

The program objectives (Reference 2) are to:

(a) Prevent detrimental non-stochastic health effects to employees.

(b) Limit detrimental stochastic health effects occurring in employees to levels as low as reasonably achievable, social and economic factors being taken into account.

(c) Provide a level of health and safety which is as good as, or better than comparable safe industries.

7.3 Responsibilities

Radiation protection in Ontario Hydro is a shared responsibility. Resources are provided for an effective program. Line managers and supervisors have demonstrated high levels of commitment in managing and supervising all aspects of the program. Individual workers apply procedures and protective equipment to perform their work safely. Health and Safety Division provides assessments of the radiation protection program and provides managers, supervisors, and workers with the information and assistance they need to carry out their responsibilities.

This system of shared responsibility in radiation protection has worked. It is likely that this positive attitude towards radiological safety has contributed to the high standard of conventional health and safety which personnel at nuclear facilities also enjoy.

7.4 The Radiation Protection Program

The following brief overview of the radiation protection program follows the main program elements contained in the policies and principles document (Reference 2).

7.4.1 General

(a) Ontario Hydro Radiation Protection Regulation

Ontario Hydro maintains a set of radiation protection regulations (Reference 4) which govern the design and operation of radiation protection programs at all nuclear facilities. This serves as a basis for the radiation protection procedures which are employed at each facility. These regulations and procedures are in compliance with all government regulations and are approved by the Atomic Energy Control Board.

(b) Expertise in Radiation Protection at Nuclear Facilities

Two groups of radiation protection specialists are employed at each nuclear facility. These are:

(1) The Radiation Control Unit - which is part of the line organization and is comprised of a radiation control supervisor and radiation control technicians. This unit provides radiation protection procedures and ensures that radiation protection equipment and materials are available when needed. It provides training in radiation protection and assists employees in the application of radiation protection procedures and equipment.

(2) The Health Physics Services unit, which is part of Health and Safety Division, is comprised of health physicists. This unit performs control and assessment functions within the radiation protection program such as approving procedures, interpreting standards and ongoing assessment of program performance and provides various services including radiation protection training, environmental monitoring and radiation dosimetry.

(c) Specialized Support in Radiation Protection

A number of organizations within Ontario Hydro, including Health and

Safety Division, Technical and Training Services Division, and Design and Development Division - Generation provide specialized support, as required.

7.4.2 Personnel Qualifications

(a) Radiation Protection Qualifications

All persons who enter the operating area of a nuclear facility are assigned a radiation protection qualification that determines their access and working rights. The qualification is valid only for the station for which it is issued.

Radiation protection qualifications are authorized jointly by line management, and Health and Safety Division, and are based on training, experience, and performance in radiation protection. The level of qualification is indicated by a colour code on the dosimetry badge:

(1) No Qualification (Red) - a person, usually a visitor, having no independent access and working rights. Must be under the direct radiological supervision of a green qualified person.

(2) Elementary Qualification (Orange) - a person having limited, designated access and working rights. To do radioactive work, the person must be under the radiological supervision of a green qualified person.

(3) Intermediate Qualification (Yellow) - a worker who has access to all parts of the facility and can independently perform radioactive work but cannot be responsible for the radiation protection of others.

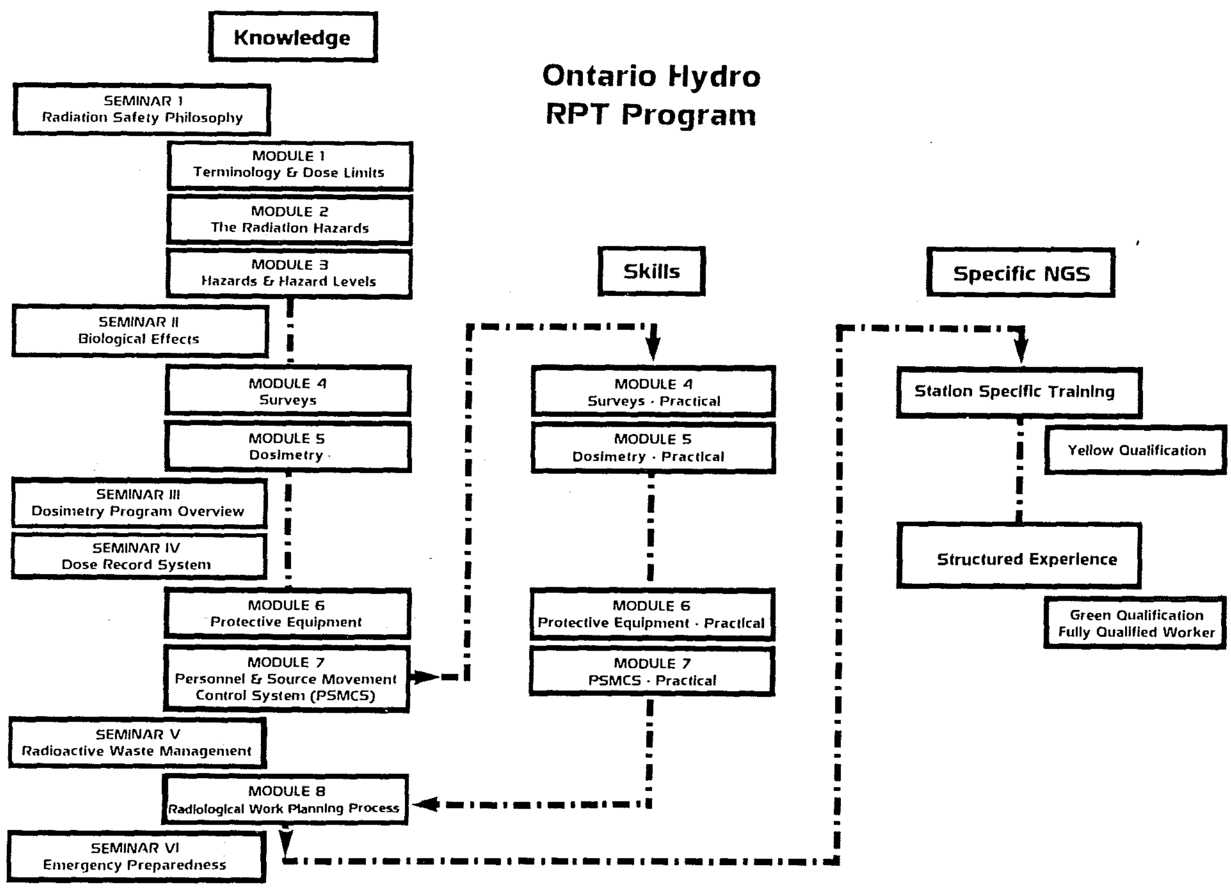
(4) Full Qualification (Green) - a worker who has full access and working rights and can be responsible for the radiation protection of others.

Intermediate and full qualifications must be maintained and are subject to periodic assessment and renewal.

(b) Radiation Protection Training

The radiation protection training is based on determinations of the knowledge and skills in radiation protection that workers would need to perform their assigned tasks safely. This training covers subjects such as radiation hazards, the biological effects of radiation, radiation surveys, dosimetry, the use of radiation protective equipment, and radiation emergency preparedness (see Figure 7-1). Health and Safety Division has overall responsibility for radiation protection training: Training in the knowledge and skills which are specific to a particular facility is conducted at the facility by the Health Physics and Radiation

FIGURE 7-1


Ontario Hydro
RPT Program
Knowledge
SEMINAR 1
Radiation Safety Philosophy
MODULE 1
Terminology & Dose Limits
MODULE 2
The Radiation Hazards
MODULE 3
Hazards & Hazard Levels
SEMINAR II
Biological Effects
MODULE 4
Surveys
MODULE 5
Dosimetry
SEMINAR III
Dosimetry Program Overview
SEMINAR IV
Dose Record System
MODULE 6
Protective Equipment
MODULE 7
Personnel & Source Movement
Control System (PSMCS)
SEMINAR V
Radioactive Waste Management
MODULE 8
Radiological Work Planning Process
SEMINAR VI
Emergency Preparedness
Skills
MODULE 4
Surveys - Practical
MODULE 5
Dosimetry - Practical
MODULE 6
Protective Equipment - Practical
MODULE 7
PSMCS - Practical
Specific NGS
Station Specific Training
Yellow Qualification
Structured Experience
Green Qualification
Fully Qualified Worker

Control units. Trainees receive a minimum of 160 hours of formal training and progress through a period of structured experience during which they apply their training to actual work situations.

7.4.3 Information Management

(a) Information on Radiation Risk

All persons who may be exposed to ionizing radiation at an Ontario Hydro nuclear facility are provided information on the risk to humans from radiation exposure. This includes risk to an unborn child when a mother is exposed during pregnancy.

(b) Radiation Dose Information System

A computerized Dose Information System (DIS) is used to collect, store, and retrieve all dosimetry information. The system provides appropriately formatted dose information reports to employees, supervisors, management, and government agencies. The system possesses considerable flexibility and can provide special information as required for the planning and control of exposures.

(c) Joint Committee on Radiation Protection

A corporate level committee, comprised of representatives from Health and Safety Division, the Production Branch, and employee organizations has been established. Its functions are to assist in the promotion of good radiation protection practices, and to review or develop revisions and/or additions to radiation protection regulations and procedures.

(d) Health and Safety Committees

Each facility maintains a joint health and safety committee comprised of representatives of management and employee organizations to address safety issues, including radiation protection.

(e) Safety Meetings

Each work unit at a facility routinely conducts safety meetings for the purpose of discussing safety issues and to provide information and training in safety related subjects. These meetings are usually held monthly.

7.4.4 Exposure Limits and Targets

(a) Legal Dose Limits

The dose limits are specified by the AECB.

(b) Dose Targets

Line managers and supervisors set annual targets for controlling individual and collective doses. These targets are usually set by work group, and are achievable, provided the work group maintains good performance in radiation protection.

7.4.5 Facility Design and Operation

(a) Facility Zoning

All areas of a facility are divided into radiological zones based on the potential that radioactive contamination may exist in that area. The purpose of zoning is to control inadvertent transfer of radioactive contamination within a facility due to movement of personnel, equipment, or material. There are monitoring requirements before movement is permitted into a zone with lower potential for contamination. Movement into clean zones is carefully controlled.

(b) Access Control Systems

Access control systems consisting of physical barriers, interlocks, alarms, and procedures are employed to ensure that there is no unauthorized access by personnel to areas where hazardous radiological conditions may exist.

(c) Ventilation Control and Containment

Ventilation systems are used to control the transfer of airborne radioactivity from contaminated areas. The flow of ventilation is from cleaner areas into the contaminated area. The air pressure in contaminated areas (containment) is kept lower than surrounding cleaner areas. Filters and dryers are used to remove contamination from air.

(d) System Design and Layout

Systems are designed with high reliability and for ease of maintenance. Systems are laid out to the extent practicable, so that those which have high radiation fields, or are major sources of airborne contamination are isolated from other systems.

(e) Material and Process Control

The material used in replacement components is controlled to preclude inadvertent entry of unwanted material such as cobalt into reactor systems. System processes and chemistry are controlled to reduce radiological hazards. It has been possible, in some cases, to use chemical decontamination methods to lower the amount of radioactivity in systems.

7.4.6 Hazard Identification and Assessment

(a) Principal Hazards

The radiation hazards found in nuclear facilities are well characterized and understood. External gamma radiation exposure and internal exposure from intake of tritium oxide account for practically all whole body dose. Beta radiation from exposed contamination accounts for some additional skin dose while hand/foot contact with contaminated surfaces causes extremity exposure. Airborne particulate contamination and loose surface contamination are potential sources of internal exposure. Radiation protection methodology addresses all of these hazards.

(b) Hazard Monitoring

Hazard monitoring and measuring methods and systems have been developed for assessing all types of radiation hazards. Routine monitoring is intended to verify that adequate control is being maintained at all times in areas where hazardous conditions could develop. Workers are trained to perform appropriate radiation and contamination surveys while performing radioactive work. Ontario Hydro has developed performance standards for radiation monitoring instruments and, in a number of cases, has found it necessary to support instrument design and development projects to meet these standards.

7.4.7 Exposure Control

(a) Radiological Work Planning

Radiological work is planned to ensure that all radiation hazards are anticipated and assessed and that appropriate protective measures are taken. The Health Physics and Radiation Control Units assist in the planning for work which has a potential for significant radiation exposure. Planning includes the development of special tools, procedures, rehearsal facilities and training as necessary.

(b) Approval of Planned Exposures

Planned exposures are estimated on the basis of known radiological conditions and exposure times. The estimated exposures and the accumulated dose each worker has already received is weighed against applicable dose limits and targets. The planned exposure for each worker is then authorized at a level of authority which is determined by the potential magnitude of the exposure.

(c) Radiation Protective Equipment

Atomic radiation workers are provided with appropriate protective clothing and respiratory protective equipment. The protective equipment is evaluated against performance standards prior to its authorization for use. Most of the protective equipment used in Ontario Hydro has been developed internally.

7.4.8 Dose Assessment

(a) Ontario Hydro's Dosimetry Program

Ontario Hydro maintains a radiation dosimetry programme (Reference 5), which is the responsibility of Health and Safety Division and which is subject to AECB approval. Appropriate dosimetric assessments for the various categories of exposure are provided, including external dosimetry for whole body, skin and extremities and bioassay analysis for tritium or other sources of internal exposure. A whole body counting service, to detect internal radiation, has been provided for all radiation workers since 1969.

(b) Dosimetry Laboratories

Dosimetry measurements involving external radiation dosimeter read-outs, bioassay analysis and in-vivo measurements are performed at three appropriately located laboratories. An important feature of laboratory operation is a quality assurance program, the results of which are audited by the AECB. This is maintained for the purpose of verifying that acceptable standards of accuracy and reliability have been attained.

7.4.9 Radioactive Material Processing, Storage and Disposal

The low and medium level radioactive waste that results from the operation of nuclear facilities is processed and stored at a waste management site located at BNPD. This site is subject to AECB licensing.

The irradiated fuel from reactors is currently stored under water in storage pools. Work is in progress to develop interim storage and long term disposal facilities for irradiated fuel.

7.4.10 Investigations

All radiation incidents and accidents are investigated to determine factors which contributed to their cause and to identify changes which are needed to ensure an acceptable standard of radiation protection for the future. As the severity or potential severity of an incident or accident increases, investigation involves a higher level of authority. Occurrences which result in serious consequences such as exposure(s) in excess of legal limits are investigated by representatives of line management and Health and Safety Division. Investigation reports are submitted to the AECB. It should be noted that radiation accidents are defined as occurrences where an acute whole body dose well in excess of dose limits is received (e.g., 0.25 Sieverts). Ontario Hydro has not experience such an accident.

7.5 Program Results

7.5.1 Summary of Collective Dose

Figure 7-2 summarizes collective dose and the corresponding dose per unit of electricity generated for the period 1975 to 1985. These data are consistent with the following trends:

(a) A significant decrease in collective dose at newer facilities. For example, the average annual collective doses at Pickering NGS A and the larger but newer Bruce NGS A for those years when each facility had four reactors in service were:

Pickering NGS A	(1973-1983):	11.94 Sv/a
Bruce NGS A	(1979-1986):	4.14 Sv/a

This decrease in collective dose at newer facilities is primarily due to improvements in systems and facility design and layout. Following early experience with high collective dose at Douglas Point, a design support function for occupational dose reduction was created to assist the design and construction organization. A formal "Occupational radiation safety engineering program" has evolved with a design review process which is actively supported by designers, radiation safety specialists in the design group and representatives from station operations and health physics staff.

(b) A decrease in collective dose vs time at each facility. This trend is most apparent at older facilities such as NPD, Douglas Point and Pickering A. For example, a significant decrease was achieved at Douglas Point as indicated by the following:

Average annual collective dose for 1970 to 1973: 14.60 Sv/a

Average annual collective dose for 1980 to 1983: 4.67 Sv/a

This trend is a measure of the effectiveness of operational improvements made by facility staff, assisted by various specialist organizations. It also demonstrates that exposures do not necessarily increase as a reactor ages.

(c) A decrease in the average individual dose per exposed worker with time. In the early 1970s, the average annual whole body dose per exposed worker was about 15 mSv with some workers in certain trade groups with doses close to the legal limit of 50 mSv. This average has been reduced to about 7 mSv/a per exposed worker (see also Figure 7-3). The risk (dose) is shared among members of a trade group provided this practise will not result in an appreciable increase in collective dose (some workers are able to perform specific tasks more efficiently with less dose because of special skills and experience).

7.5.2 Comparison with Other Countries

Data on parameters related to radiation protection performance are available (Reference 7) for a number of countries. Figures 7-3, 7-4 and 7-5 summarize some key parameters from the most recently available information and who that Ontario Hydro enjoys generally favourable experience in collective dose and dose per unit of electricity generated. The annual whole body dose for each exposed worker (currently approximately 7 mSv) is comparable with the average for US stations but higher than for some other countries. This is consistent with the fact that the number of exposed workers per reactor unit in service in Ontario Hydro is significantly lower than these countries. This is in line with the objective of operating the nuclear stations with a normal staff complement and avoiding the use of large numbers of attached workers.

7.5.3 Epidemiological Study

Ontario Hydro sponsors an on-going epidemiological study for cause of death for male employees on the radiation does registry (1970 - present) and for male employees according to current, or (in the case of pensioners) the last place of work - Nuclear Generating Station, Thermal Generating Station, or other employees of Ontario Hydro. Causes of death are neoplasms, cardiovascular, accidents, and all other causes. Table 7-3 summarizes results from the most recent report (Reference 8). The standardized mortality ratios (SMR"s) can provide warning of a possible cause/effect mechanism. It should be noted that although the nuclear group enjoys a very favourable experience in terms of its SMR, there has probably not been a long enough "latent period" (leukemia being the notable exception) for radiation induced cancers to appear.

7.5.4 Conventional Health and Safety

Table 7-4 presents the fatality rate experienced by various industries in Ontario (Reference 9). Ontario Hydro has experienced no fatal accident from the operation of a nuclear facility following greater than 10^8 man-hours of operation.

7.5.5 Conclusion

The standard of employee radiological health and safety in Ontario Hydro nuclear facilities is acceptable in comparison to standards of other Ontario industries and nuclear facilities in other countries. The high level of commitment throughout the corporation which is responsible for this achievement is firmly established and promises to ensure that all future standards for acceptable health and safety will be attained.

TABLE 7-3

Cumulative Mortality Experience
By Location: 1970 - 1985 (8)

		Nuclear	Thermal	Other	Total
NEOPLASMS	Obs	17	41	648	706
(Cancers)	Exp	24.86	39.11	734.66	798.63
	SMR	68	105	88	88
CIRCULATION	Obs	20	58	1453	1531
(Heart,	Exp	40.59	75.43	1635.51	1751.53
stroke, etc.)	SMR	49	77	89	87
ACCIDENTS	Obs	34	15	182	231
(all types)	Exp	43.40	28.24	245.82	317.46
	SMR	78	53	74	73
ALL OTHER	Obs	5	18	368	391
CAUSES	Exp	20.82	29.23	568.28	618.33
	SMR	(24)	62	65	63
TOTAL	Obs	76	132	2652*	2860*
(all causes)	Exp	129.67	172.01	3184.27	3485.95
	SMR	59	77	83	82

Accidents include non-work related facilities.

Observed (Obs) and Expected (Exp) deaths, and Standardized Mortality Ratios (SMRs) in all males employees of Ontario Hydro (active pensionable and pensioned, combined) according to place of work.

Expected figures are aged-adjusted by five-year age-groups and are based on general male mortality rates in Ontario during the intercensus periods 1971 - 1976 and 1976 - 1981.

SMRs express observed as a percentage of expected and have been put in parentheses where based on less than ten observed deaths.

*Note: These total are larger than the sum of the sub-categories, because one cause of death was not yet known at the time of printing.

TABLE 7-4

Fatal Accidents for Major Industries in Ontario

(1981 - 1984)

Schedule 1	Number of Fatalities*	Frequency (10^8 man-hours)
Agriculture	38	13.5
Forestry	30	40
Mining, Quarrying & Oil Wells	88	32.8
Manufacturing	207	2.5
Construction	128	13.3
Transportation, Communication and Other Utilities	89	15
Trade	38	1
Services	43	1.3
Public Administration	12	3.8
Other (Fishing)	5	111.8
Overall	688	3.3

*Fatalities from occupational disease and trauma.

REFERENCES

(1) The Organization and Operation of Ontario Hydro Series 1, Section. 4-5/Corporate policies 85-07-15.

(2) Ontario Hydro Radiation Protection Policies and Principles.

(3) Atomic Energy Control Regulations - regulations made pursuant to the Atomic Energy Control Act.

(4) Ontario Hydro Radiation Protection Regulations, Part 1.

(5) The Ontario Hydro Radiation Dosimetry Programme, (in three volumes) by D.W. Whillans, D.A. Agnew and M.L. Walsh.

(6) NGD Occupational Radiation Dose Summary, prepared by the Radiation Dose Reduction/Occupational Safety Section of RMEP [available from 1962 through 1986].

(7) Comparison of Ontario Hydro Occupational Radiation Dose Performance to Other Utilities/Countries Using Nuclear Power, October, 1986, A. Steer, RMEP.

(8) Ontario Hydro Mortality, 1970 - 1985, by T.W. Anderson, Professor and Head, Department of Health Care and Epidemiology. Faculty of Medicine. University of British Columbia.

(9) Advisory Council on Occupational Health and Occupational Safety, Eighth Annual Report, April 1, 1985 to March 31, 1986, Volume 1.

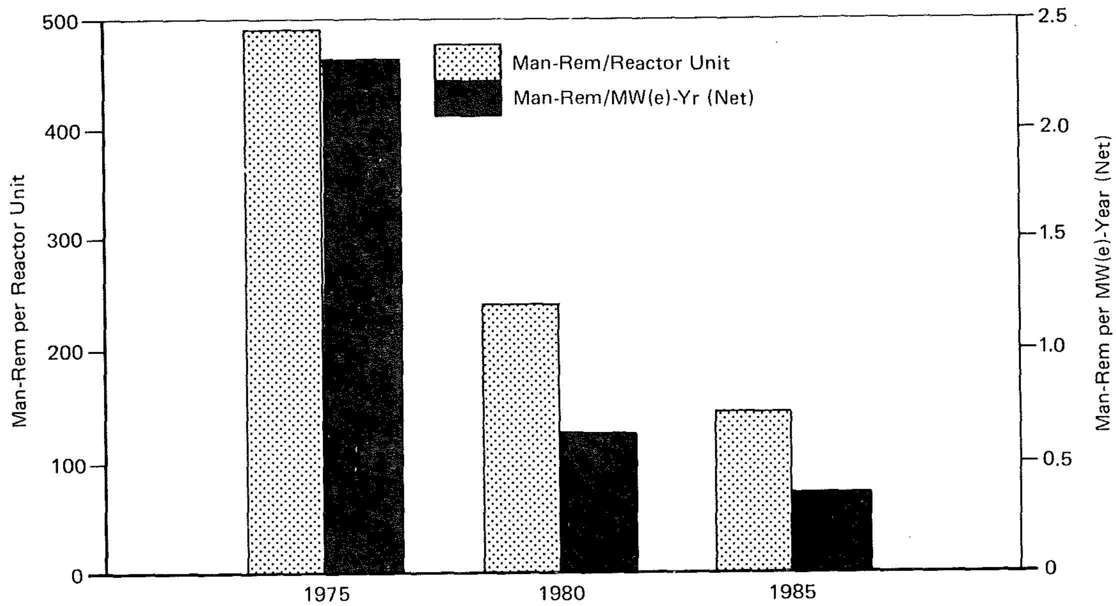


FIGURE 7-2
Collective Dose Performance

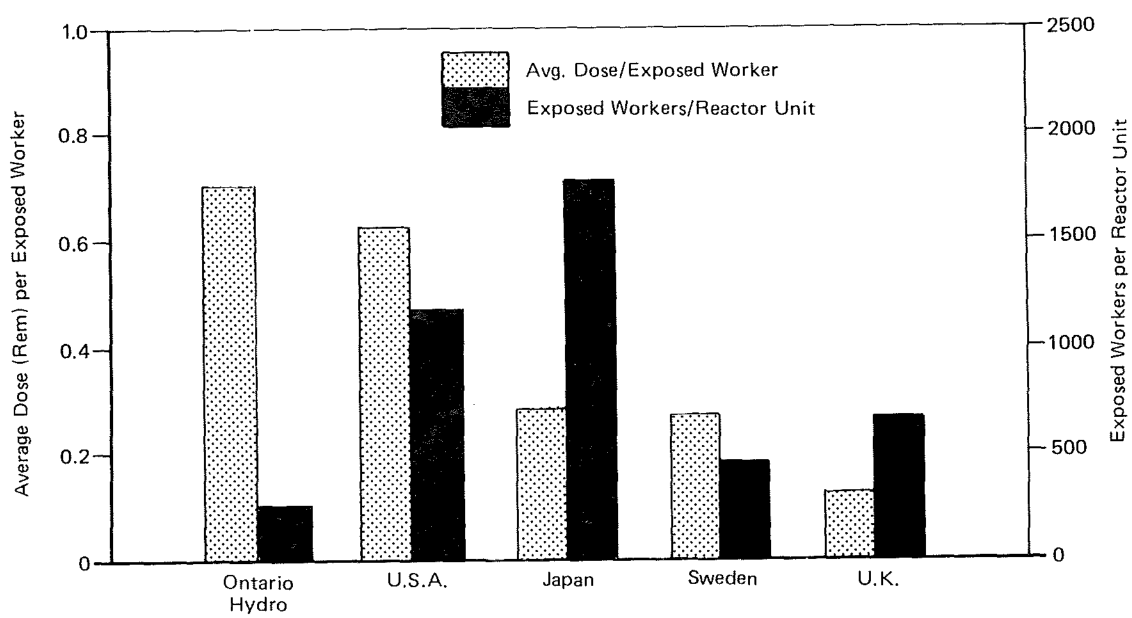


FIGURE 7-3
Exposed Workers and Average Doses
(5 Year Average Doses)

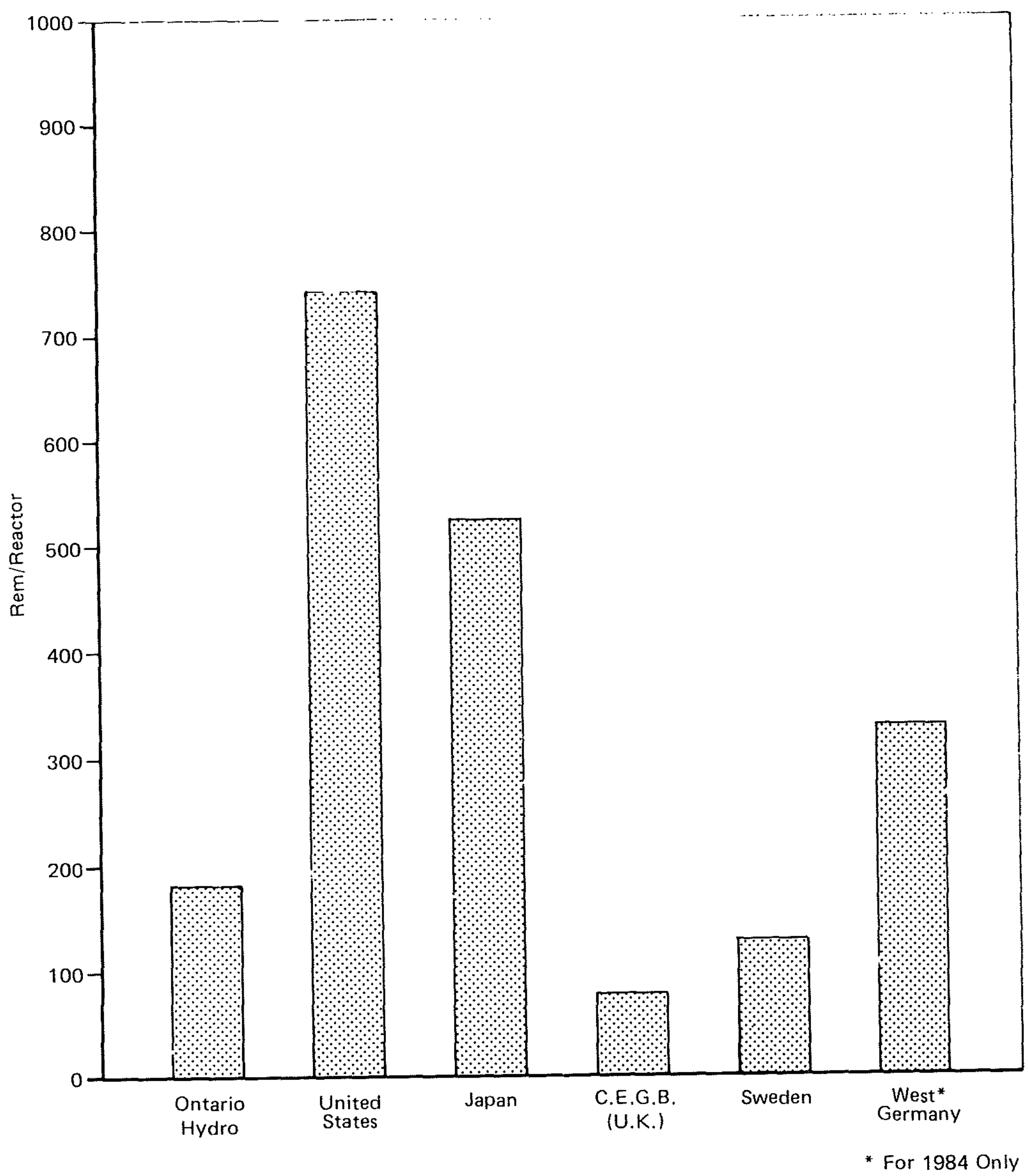


FIGURE 7-4
Annual Dose (Rem) per Reactor, 5-Year Average
1980 – 1984

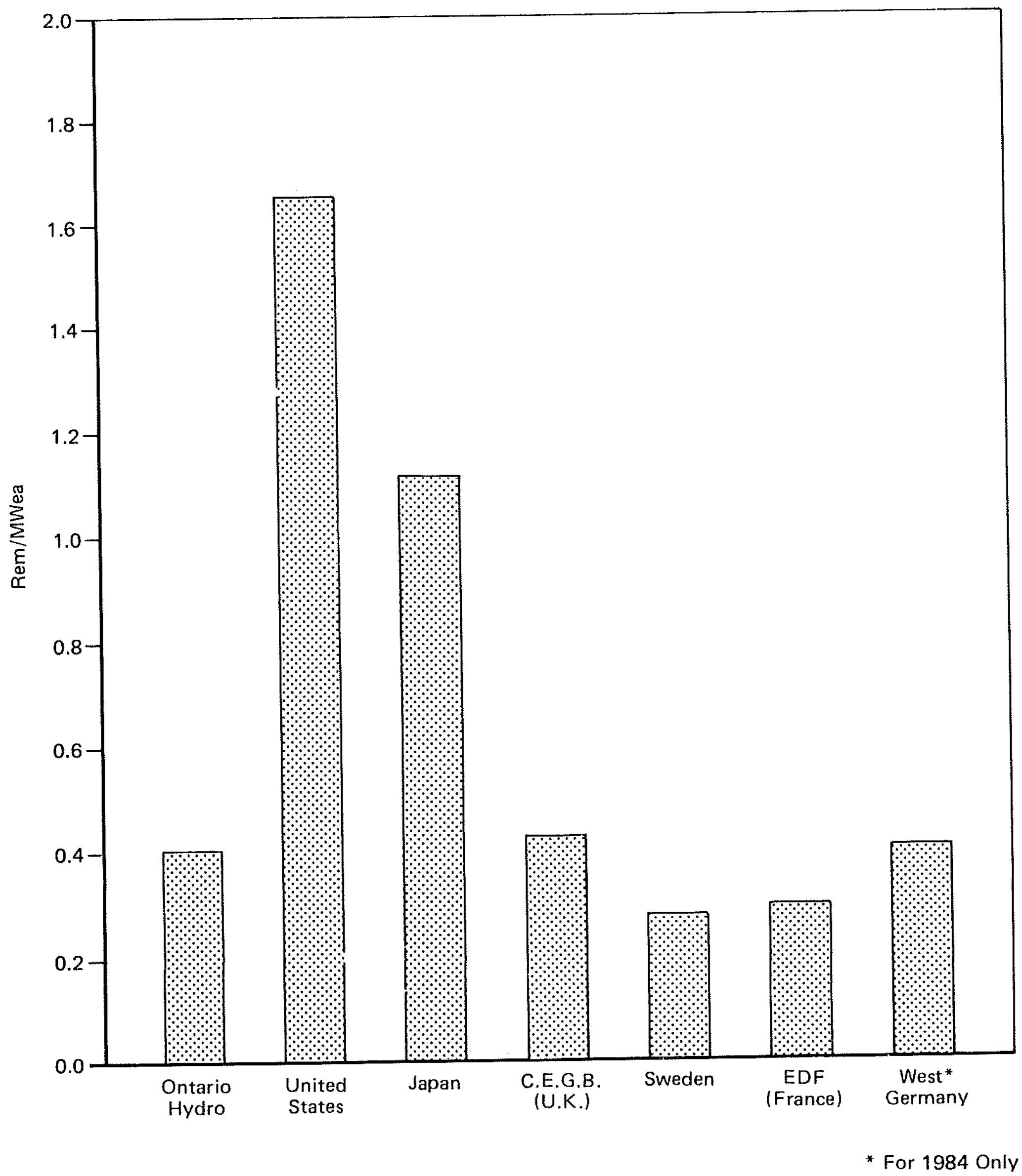


FIGURE 7-5
Dose per Gross Megawatt-Year (Electrical)
5-Year Average
1980 – 1984

Operating Results – Safety Performance and Environmental Emissions

8.0 OPERATING RESULTS - SAFETY PERFORMANCE AND ENVIRONMENTAL EMISSIONS

This section provides information on Ontario Hydro's achieved performance in controlling the public risk arising from the operation of nuclear power stations.

There is a potential risk that radioactive material could be released as a result of a major accident. The degree of protection which has been provided against such accidental releases is discussed in Section 8.1.

The public may also be exposed to radiation as a result of radioactive material being emitted from the station to the environment, under normal operating conditions. Under these conditions small leaks and spills occur which may cause emissions to the environment before remedial action can be taken. These emissions are controlled to levels which are "as low as reasonably achievable, social and economic factors being taken into account" (the internationally accepted ALARA principle). A measure of success of these control actions is shown in Section 8.2.

8.1 Reactor Safety Operating Results

The evaluation of risk to the public from a possible accident at a nuclear station requires a consideration of the following:

(a) the frequency of the accident, and
(b) the consequences of the accident.

Reduced risk can be achieved by reducing the frequency (probability) of an accident and/or reducing the consequences.

The public can potentially be subjected to an acute radiation exposure only if a serious failure occurs in one of the normally operating plant systems (process failure). The consequences of a process failure can be compounded if a safety system provided to protect against that process failure also fails.

During the design of a nuclear station, reliability targets are set for process systems and safety systems. These targets are chosen such that the objective of negligible public risk will be achieved. During operation, a safety management program (described in Section 5) is implemented to ensure that these reliability targets are met.

The effectiveness of this management program in ensuring negligible public risk can be judged by observing the performance of safety related systems.

The two main indicators of safety performance are:

(a) the frequency with which process system failures occur, and

(b) the availability of safety systems to protect against or mitigate the consequences of process system failures.

The following sections present a summary of these two indicators for Ontario Hydro nuclear generating stations over the period 1980 to 1986. Changes in reporting practices, modifications to system equipment and procedures, and new analyses result in older data becoming less relevant to current performance. The period 1980 to 1986 was chosen to minimize this effect, while still being long enough to enable trends to be discerned. Older data are presently being reviewed to make them compatible with current reporting standards. Statistics for Douglas Point Generating Station have not been included as this station is no longer operating.

Further details of the performance of safety related systems can be found in the various reports listed under References. The most comprehensive information is contained in each station's Quarterly Technical Reports.

8.1.1 Process System Failures

All significant events involving failures in process systems, whether caused by equipment failure or human error, are reviewed and categorized according to the nature and severity of the fault. The classification scheme uses five categories of faults, labelled Type A, B, C, D, E. Type A and B faults are those considered most serious and with the greatest implications for safety. Type C and D faults are less serious events which did not, in themselves cause an immediate threat. Type E faults are benign or tended to put the system in a safe state.

The definition of Type A faults is:

"A fault which raised fuel temperatures and which would have caused significant fuel failures in the absence of Special Safety System action."

The term "serious process failure" is synonymous with a Type A fault. The AECB licensing criteria limit the frequency of all such failures to less than 1 per 3 years.

A Type B fault is a "near miss" Type A - ie, the fault would have been classified as a Type A if some unpredictable factors (eg, reactor power level) had been different.

Table 8-1 below gives the total number of Type A and Type B faults reported for each NGD station over the period 1980 to 1986.

Table 8-1

	Pickering A	Bruce A	Pickering B	Bruce B
Type A	1	0	0	0
Type B	8	10	1	0
Unit-years experience in review period	13	20	10	6

From Table 8-1, the frequency of serious process failures - Type A faults - has been very low and is certainly well within the licensing criterion of 1 per 3 years per unit.

Only one Type A fault has occurred in this period: a loss of regulation event at Pickering NGS-A. (A "loss of regulation" is an uncontrolled increase in reactor power which requires operation of the shutdown system in order to prevent fuel failures.) In this particular event, a failure in the control computers resulted in an increase in reactor power. The shutdown system acted automatically as designed to safely shut the reactor down. Modifications have been made to the control computers to prevent recurrence of this particular fault.

Although performance prior to 1982 was not included in Table 8-1, as being less relevant to current performance, it should be noted that performance was not as good in the early years. A total of 9 Type A faults occurred prior to 1982, the majority of which were Loss of Regulation events at Pickering NGS-A. As a result of this poor performance, a thorough review of the reactor control system at Pickering, was undertaken. Improvements to the system were made at Pickering NGS-A, and at the other stations (including those being designed or constructed), resulting in the significantly improved performance over recent years.

Various Type B faults have occurred - at a frequency less than 1 per reactor-year. No specific standard has been set for the frequency of these faults, however each fault is thoroughly reviewed to determine if corrective actions are required to prevent recurrence.

Type A and B failures are automatically reviewed by the Nuclear Integrity Review Committee (NIRC). Other process failures are scrutinized by technical staff and referred to NIRC if they are considered to be of major reactor safety significance. An example of a failure of this nature, which did have major implications is the pressure tube rupture event at Pickering NGS-A in August 1983. In that incident the operating staff coped with the event in a competent fashion, Special Safety System action was not required, and no release of radioactivity resulted. This event was originally classified as a Type B failure, due to its apparent severity, but was later reclassified as a Type D because Special Safety System action was not necessary. Review of this event resulted in the replacement of all pressure tubes in Pickering NGS-A Units 1 and 2.

8.1.2 Special Safety System Performance

The fraction of time that Special Safety Systems are unavailable must be small such that the predicted frequency of "dual failures" (a serious process failure combined with failure of a Special Safety System) stays within the AECB licensing criteria.

The major indicator of the past performance of Special Safety systems is "Actual Past Unavailability," which is defined as the time fraction that the overall system was known to be not fully available (ie, not fully capable of meeting its design intent) during the past year. It should be noted that this indicator provides a very conservative measure of performance: faults which marginally reduce the effectiveness of the system for one particular type of process failure are included in this measure, even though the system may still be capable of providing adequate protection for most events. Also, this index is susceptible to large statistical fluctuations from year to year.

In general, this indicator is not corrected retroactively for deficiencies in the system design which may be later discovered, since it is a measure of the operability of the as-installed system. The index is adjusted, however, to account for faults which may be reclassified from their initial conservatively designated category if later analysis and review indicates they are less severe. Two or three incidents are typically under review at any given time.

A supplementary measure of system performance is "System Inoperability." This indicator is defined as the fraction of time that a system was fully incapable of providing protection for the events with which it was designed to cope. As this measure does not include partial system failures it is a nonconservative measure of performance. However, when used in conjunction with Actual Past Unavailability it is useful in distinguishing between major system faults which definitely affect public risk and faults which may, in reality, represent only an erosion of the conservative assumptions used in the plant safety analysis.

The performance of Special Safety Systems at each station over the period 1982 to 1986 in terms of "Actual Past Unavailability" and "System Inoperability", summarized in Figures 8-1 to 8-4. The data are presented as they were reported based on the test results for a given year.

It should be noted that the unavailability targets quoted for each system are targets (ie, tools to help in judging performance) rather than hard limits because of the following:

- the significance of not meeting a target depends heavily on the success and failure criteria used. Use of very strict criteria, with a large degree of conservatism, can result in systems not meeting the target because of one or more marginal deficiencies even though the effect on overall risk is negligible;

- the statistical uncertainties involved can be quite large.

Other means of assessing safety system performance which are used routinely are "Predicted Future Unavailability" and component failure rate trending. Predicted Future Unavailability is a calculated reliability index which combines all relevant component failure data on the system of interest to provide an estimate of long-term average system performance.

The following sections review the performance data for each system in turn.

8.1.2.1 Shutdown Systems (Table 8.2)

The unavailability data reported for shutdown systems for each station (Figure 8-1 to 8-4) show that Actual Past Unavailability has exceeded target infrequently. Occasional problems have been experienced, which required design modifications or procedural changes, but no major deficiencies are evident. The Figures also show that very little Inoperability has been associated with shutdown systems.

8.1.2.2 Emergency Coolant Injection Systems

The reported unavailability data for Emergency Coolant Injection Systems (ECIS) shows a number of occasions when targets have not been met. Problems have been experienced with the ECIS at both Bruce NGS-A and Pickering NGS-A, with the systems frequently being over target. However, numerous improvements have been made to these systems resulting in improved performance in recent years. The entire ECIS at Bruce NGS-A has now been upgraded with a system similar to that installed at Bruce NGS-B. Pickering NGS-A is also having new ECIS equipment installed.

8.1.2.3 Containment Systems

The reported unavailability data for Containment systems for each station show that the unavailability targets have been met most years at most stations.

Bruce NGS-A has had a recurring problem in recent years which has resulted in Actual Past Unavailability targets being exceeded. The problem involved dampers in the containment ventilation system leading to a delay in the automatic isolation of containment in the event of certain accidents. Procedural changes have been made, and a design modification is planned in order to prevent future occurrences of this fault.

A problem was also discovered during testing of the containment system at Bruce NGS-A in May 1987. A check valve in a system associated with the Vacuum Building was found failed. The significance of this failure is still under review, and existence of the fault is not reflected in Figure 8-3.

8.2 Chronic Emissions to the Environment

Environmental protection is one of the five key effectiveness areas used by Nuclear Generation Division (NGD) to measure performance. Standards of environmental protection performance have been established for NGD based upon regulatory requirements or guidelines. Data to demonstrate compliance are collected and reported annually in the "NGD Annual Environmental Summary" (Reference 10).

In 1986 all NGD environmental protection performance standards except for conventional oil spills were met, ensuring that the impact of the operation of nuclear facilities on the environment was minimal.

8.2.1 Radioactive Emissions

Radiological dose limits for members of the public have been recommended by the International Commission on Radiological Protection and adopted by the AECB as the regulatory limits (Reference 11).

Possible radionuclide emissions from nuclear facilities have been identified. Pathways by which they may come into contact with those members of the public most likely to be exposed to a particular radionuclide group, have been analyzed.

Using this information and making conservative assumptions, derived emission limits (DELs) have been calculated for each Nuclear Generating Station (NGS) to restrict the radiological dose to members of the public to less than the regulatory limit. DELs for significant radionuclide categories which have been approved by the AECB for each operating Ontario Hydro facility are presented in Figures 8-5 to 8-8.

In the early 1970s, Ontario Hydro adopted an operating target for radiological emissions of 1% or less of each DEL.

8.2.2 Conventional Emissions to the Environment

Conventional emissions to the environment have also been identified and are monitored. Performance standards have been set based on Provincial Environmental Legislation or Guidelines. The following are the parameters which are monitored.

Thermal Emissions

Thermal emissions from all facilities were monitored and were normally below the hourly operating targets for effluent temperature rise and effluent temperature.

Chemical Emissions

Continuous ambient monitoring of the environment at seven locations around the BNPD site demonstrated that hydrogen sulphide (H_2S) and sulphur dioxide (SO_2) levels remained very low. Standards of performance for both were met.

The hourly loading and concentration limits for H_2S to water at BHWP were both exceeded on one occasion during 1986.

A major spill of triarylphosphate ester (TAP ester) to Lake Huron occurred at BNGS-A. Two oil spills also occurred at BNGS-A.

Conventional Waste Management

Management of conventional waste materials received more attention during 1986 because of changing regulatory requirements.

All NGS's were registered as waste generators under Reg. 309 of the Environmental Protection Act. The information supplied to the Ministry of the Environment (MOE) will facilitate control of wastes from generation through to disposal.

Polychlorinated Biphenyls (PCB) continued to be stored at each station in registered and approved storage sites conforming to requirements in Reg. 11/82 under the Environmental Protection Act. A BNPD site PCB storage location was commissioned in 1986. It will be used for Bruce Site Construction and Operations PCB waste.

Odour

There were 4 incidents at Bruce Heavy Water Plant (BHWP) that generated odour complaints from local residents.

Noise

Three incidents occurred that generated noise complaints from the general public; two at BNPD, the third at PNGS.

Fish Loss

Total fish loss at water intakes was 98 000 kg. Most fish loss occurred at PNGS where 80 000 kg of alewife and 9 800 kg of gizzard shad were collected. The amounts of alewife, salmon and trout removed from Lake Ontario by PNGS represented a very small fraction of total removal of these fish from the lake (including sport and commercial fishing). Coarse fish (alewife, gizzard shad, etc), accounted for 98% of total fish loss at all intakes, consistent with the results of previous years.

References

The following is a list of the principal reports which contain further information on public safety performance.

(1) NPD Quarterly Technical Reports.

(2) Pickering NGS-A Quarterly Technical Reports.

(3) Bruce NGS-A Quarterly Technical Reports.

(4) Pickering NGS-B Quarterly Technical Reports.

(5) Bruce NGS-B Quarterly Technical Reports.

(6) Report NGD-11 "Performance Review" - (Produced Annually).

(7) Report NGD-9 "Ontario Hydro CANDU Operating Experience" (Revised Annually).

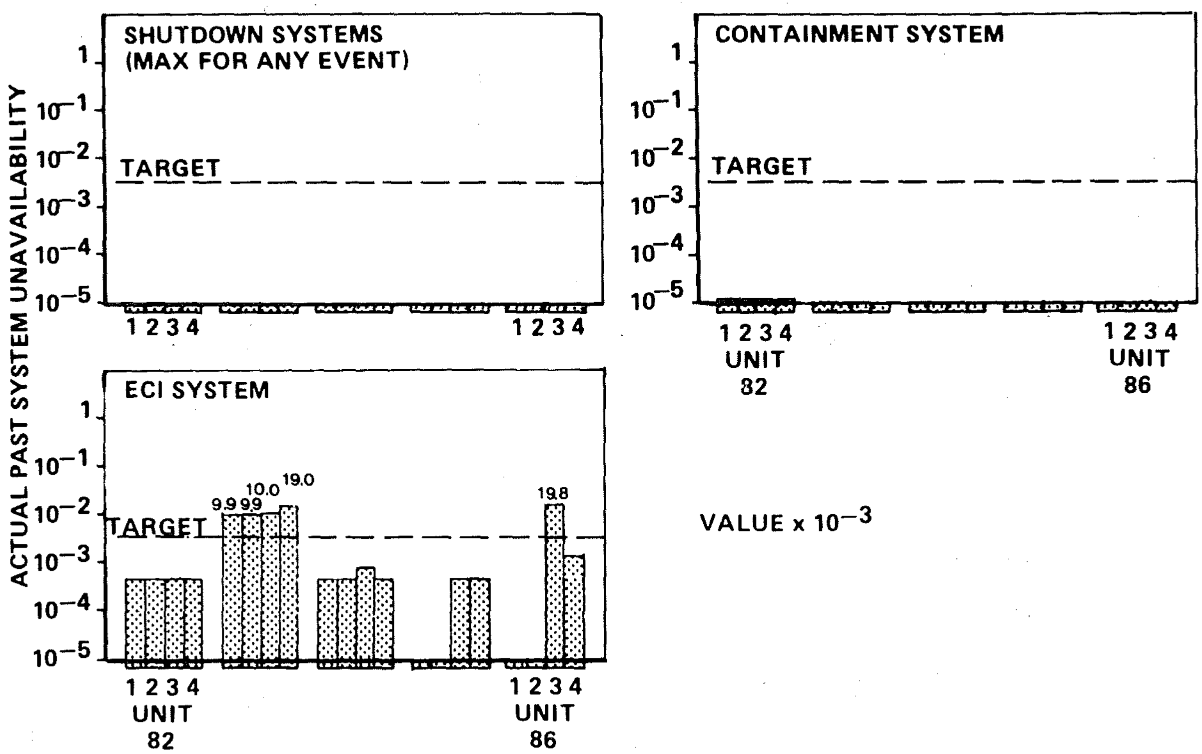
(8) Nuclear Integrity Review Committee Annual Reports.

(9) Report RMEP-IR 9053 - "Annual Review of Significant Event Reports in the Nuclear Generation Division."

(10) Report RMEP-IR-07000-13", NGD Annual Environmental Summary", 1977 to 1986.

(11) Ontario Hydro Radiation Protection Regulations, Part 1.

PICKERING NGS-A
SAFETY SYSTEM PERFORMANCE TRENDS – UNAVAILABILITY



PICKERING NGS-A
SAFETY SYSTEM PERFORMANCE TRENDS – INOPERABILITY

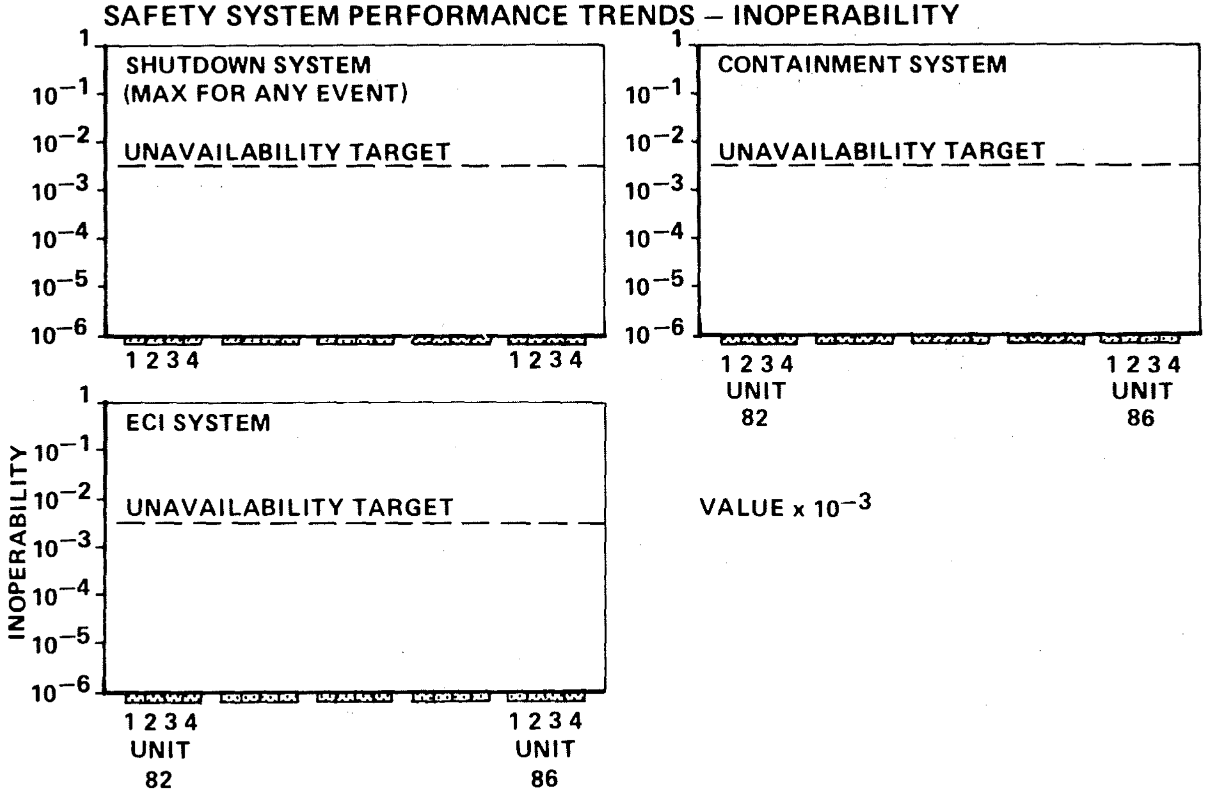
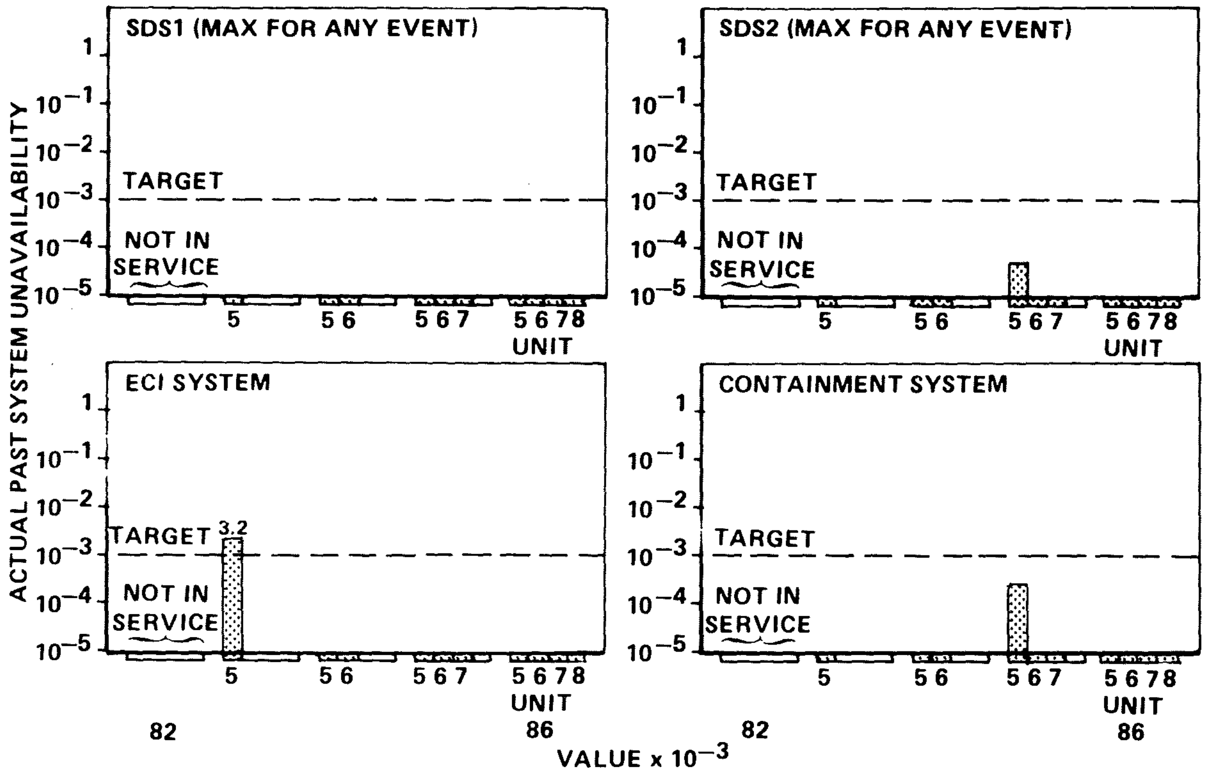


FIGURE 8-1

PICKERING NGS-B
SAFETY SYSTEM PERFORMANCE TRENDS – UNAVAILABILITY



PICKERING NGS-B
SAFETY SYSTEM PERFORMANCE TRENDS – INOPERABILITY

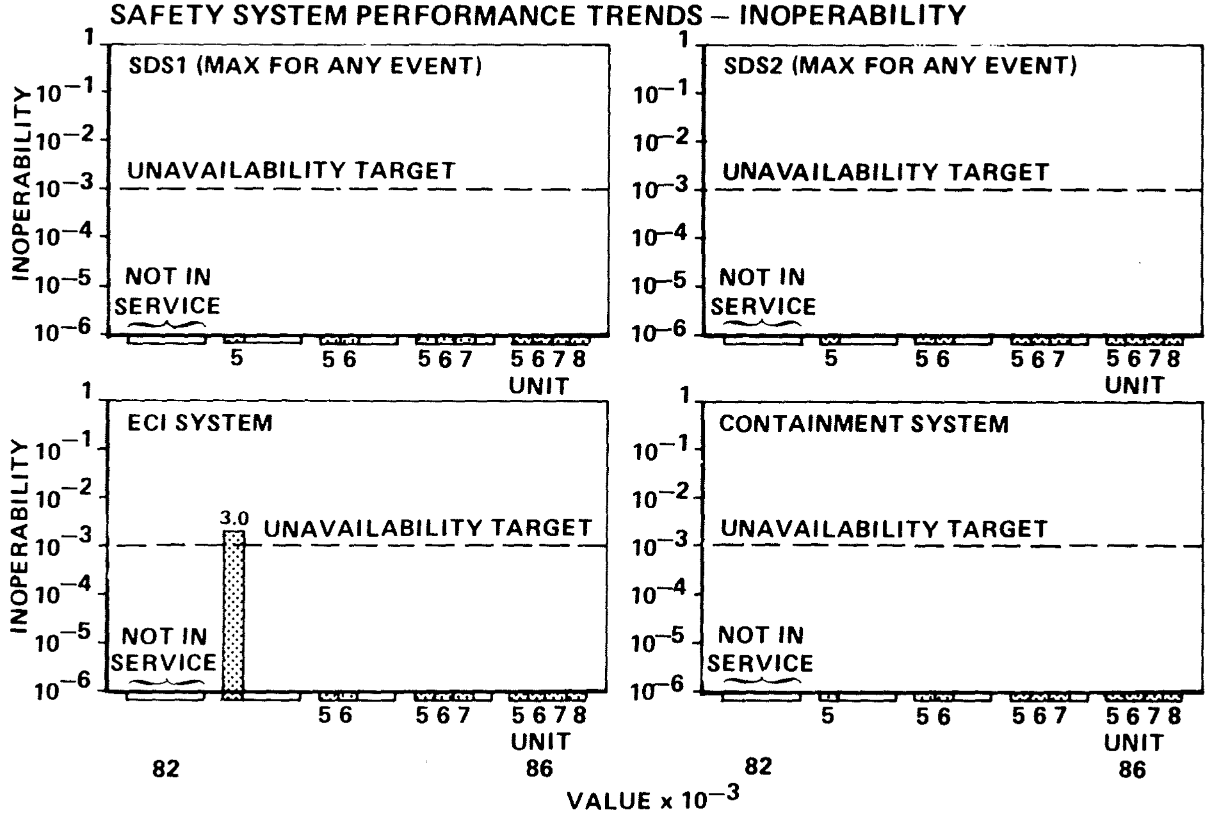
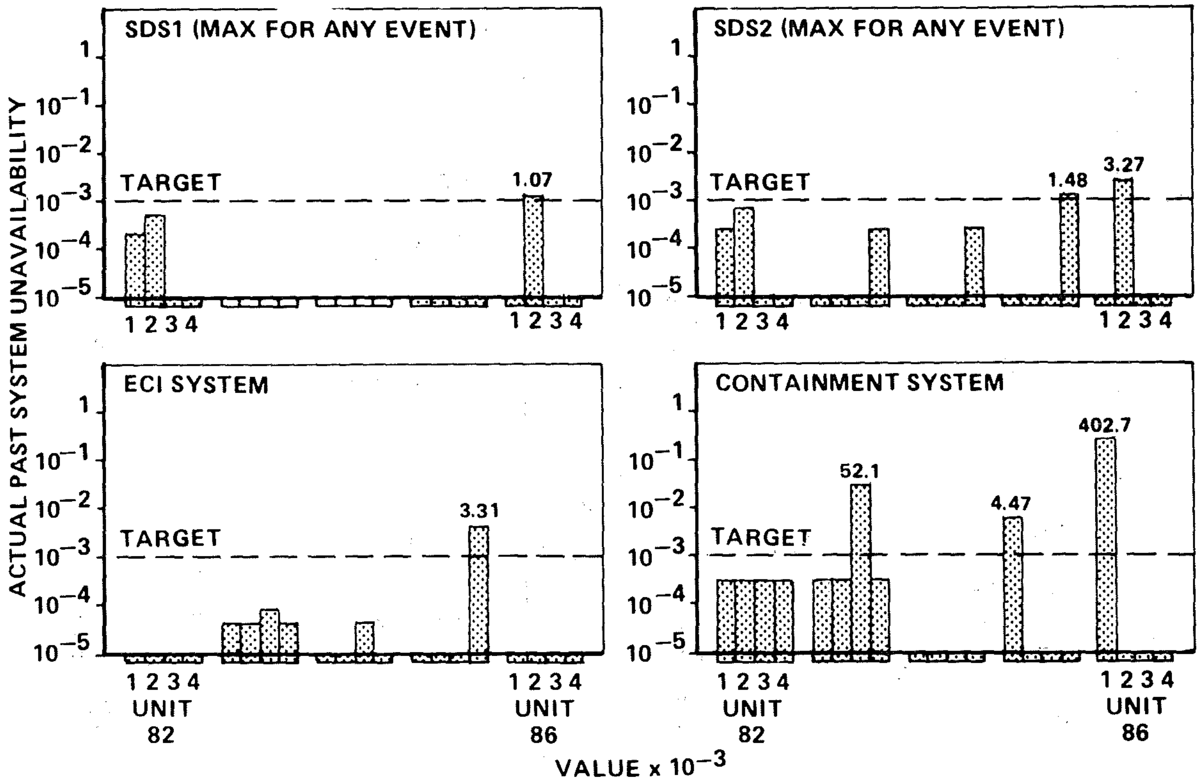


FIGURE 8-2

BRUCE NGS-A
SAFETY SYSTEM PERFORMANCE TRENDS – UNAVAILABILITY



BRUCE NGS-A
SAFETY SYSTEM PERFORMANCE TRENDS – INOPERABILITY

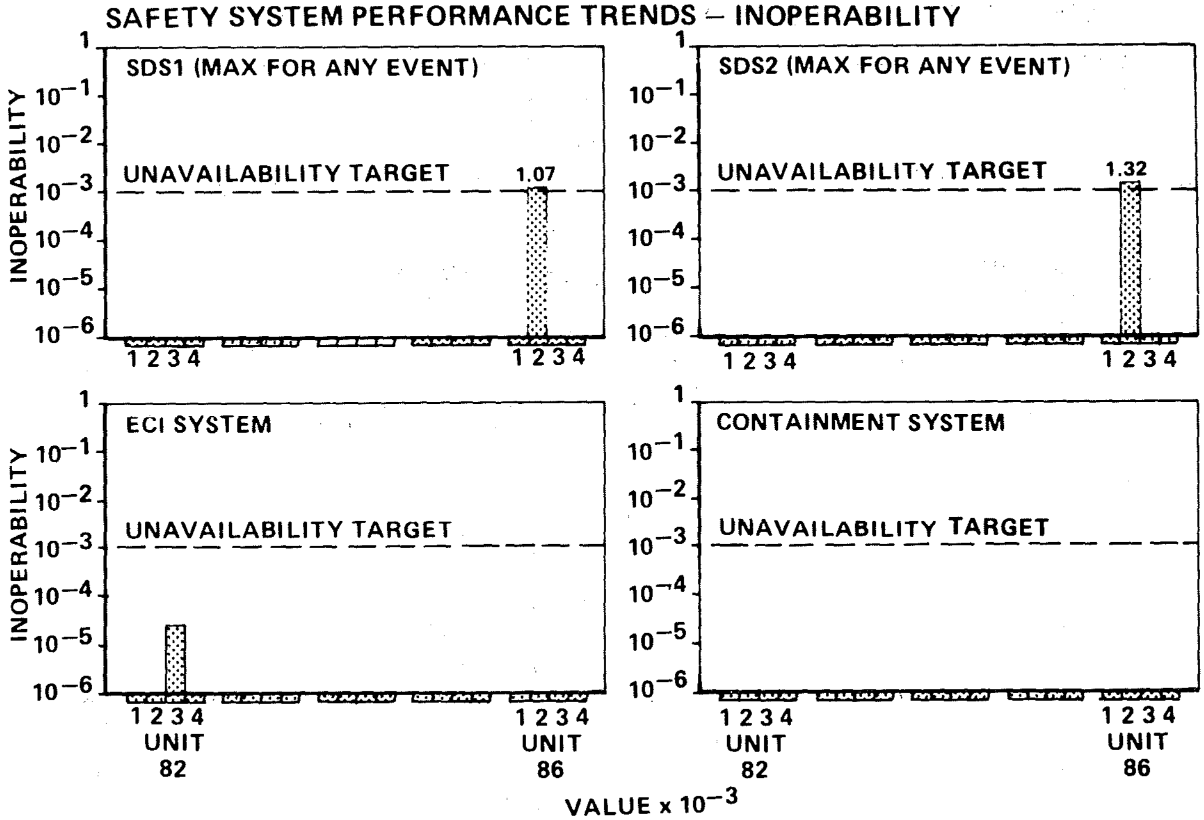
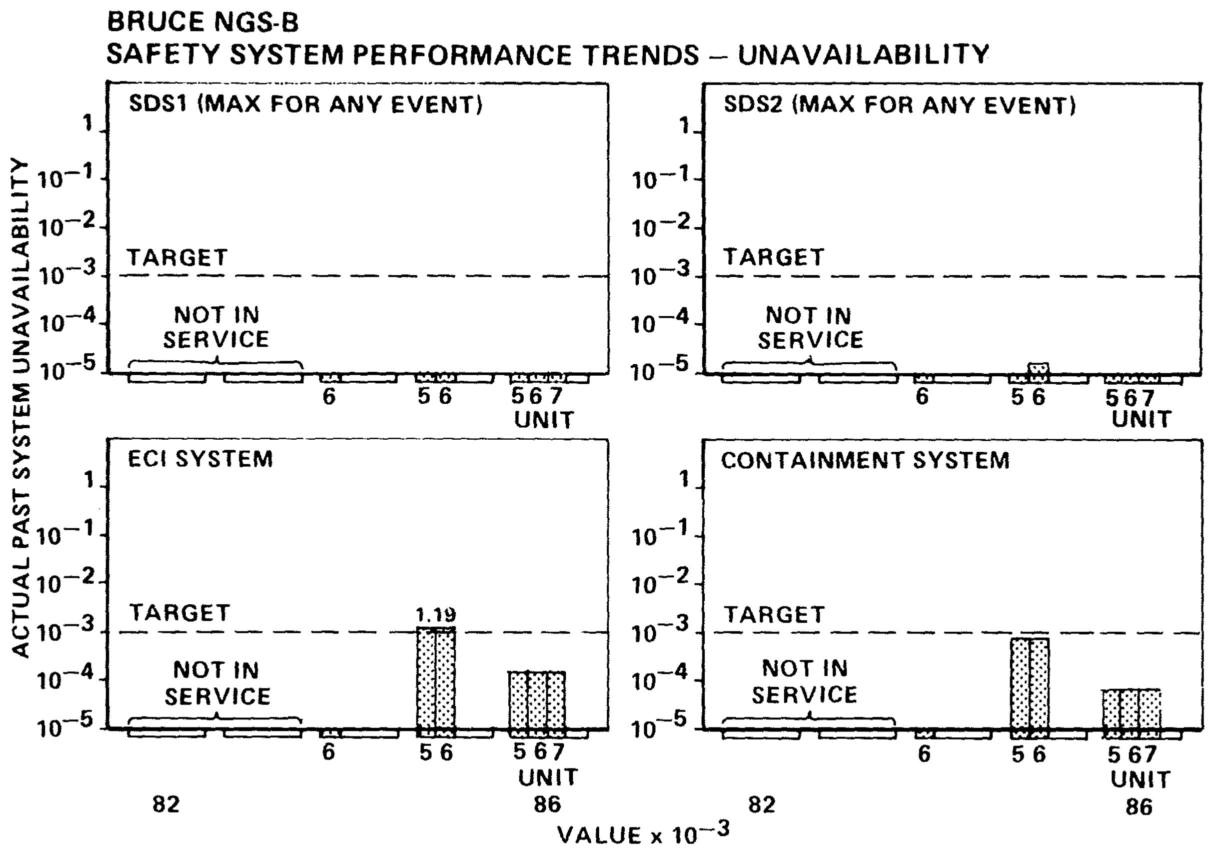


FIGURE 8-3


BRUCE NGS-B
SAFETY SYSTEM PERFORMANCE TRENDS – UNAVAILABILITY
ACTUAL PAST SYSTEM UNAVAILABILITY
SDS1 (MAX FOR ANY EVENT)
SDS2 (MAX FOR ANY EVENT)
ECI SYSTEM
CONTAINMENT SYSTEM
TARGET
NOT IN SERVICE
1.19
UNIT
82
86
VALUE x 10^{-3}

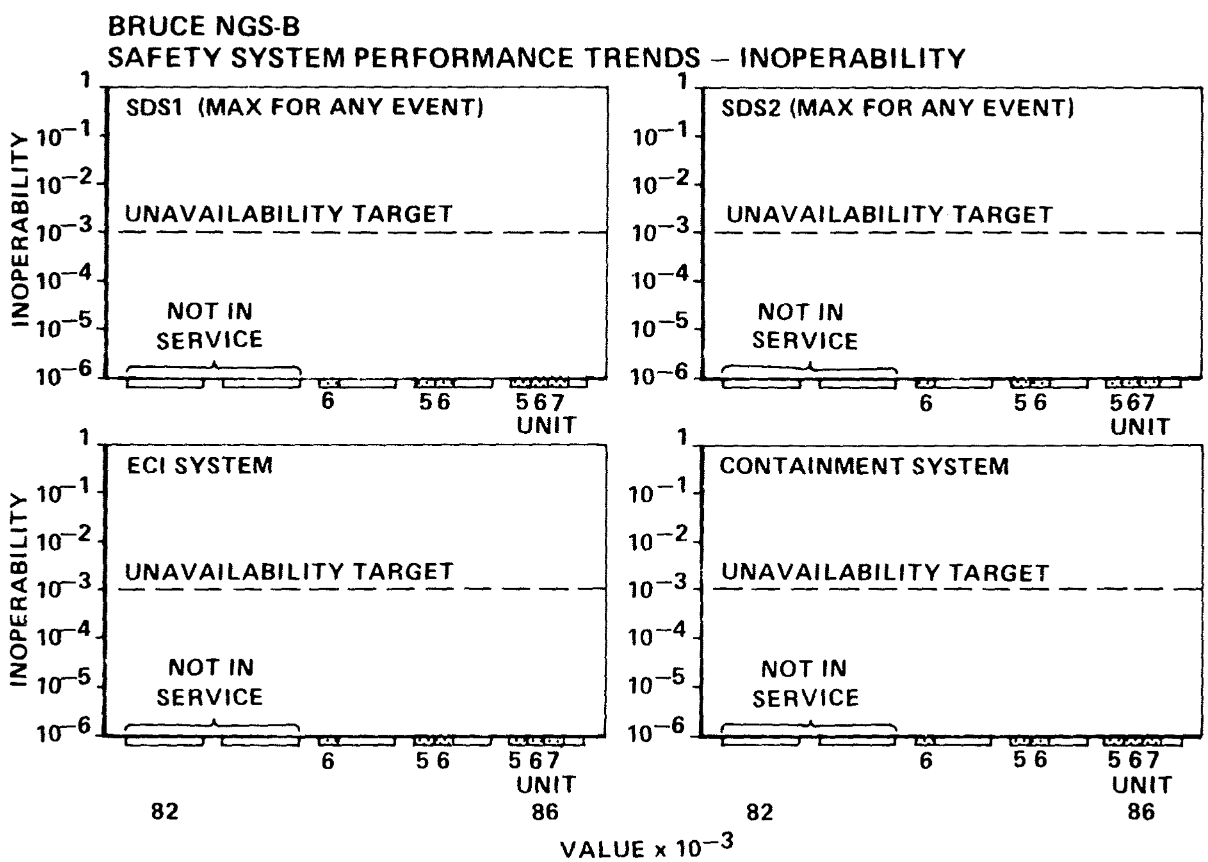
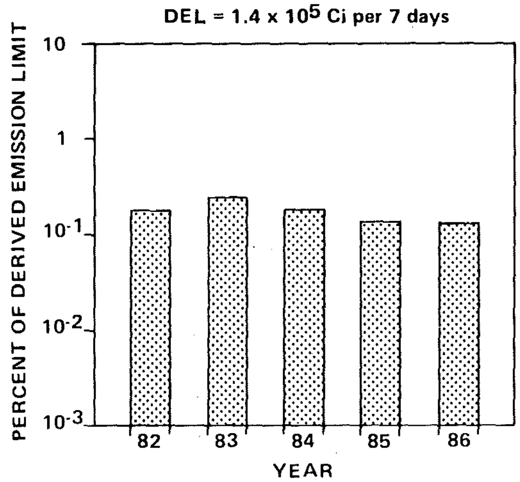

BRUCE NGS-B
SAFETY SYSTEM PERFORMANCE TRENDS – INOPERABILITY
INOPERABILITY
SDS1 (MAX FOR ANY EVENT)
SDS2 (MAX FOR ANY EVENT)
ECI SYSTEM
CONTAINMENT SYSTEM
UNAVAILABILITY TARGET
NOT IN SERVICE
UNIT
82
86
VALUE x 10^{-3}

FIGURE 8-4

FIGURE 8-5
PICKERING NGS-A RADIOACTIVE EMISSION TREND
AIR

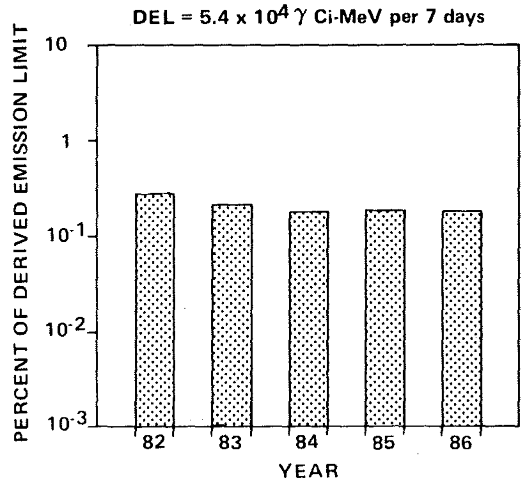
TRITIUM

DEL = 1.4 x 10^5 Ci per 7 days



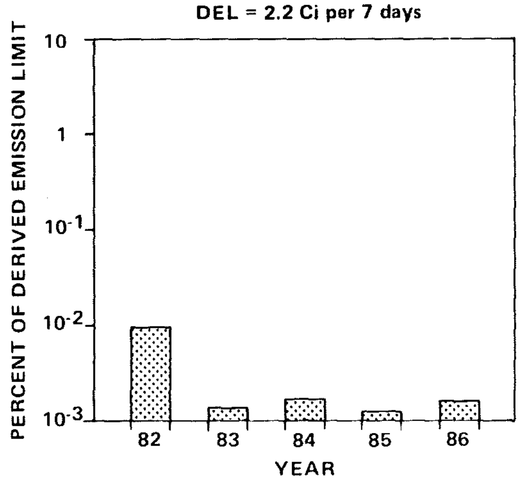
NOBLE GASES

DEL = 5.4 x 10^4 γ Ci-MeV per 7 days



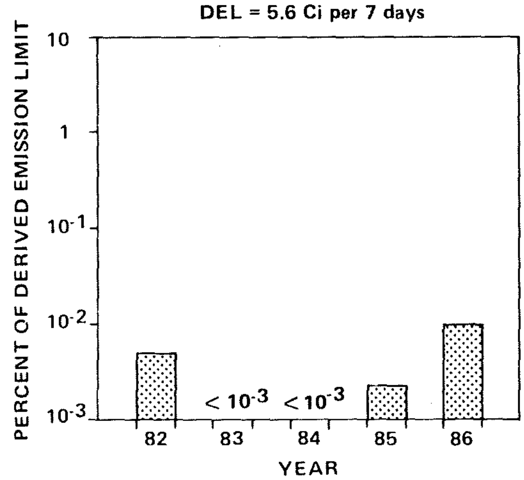
IODINE-131

DEL = 2.2 Ci per 7 days



PARTICULATE

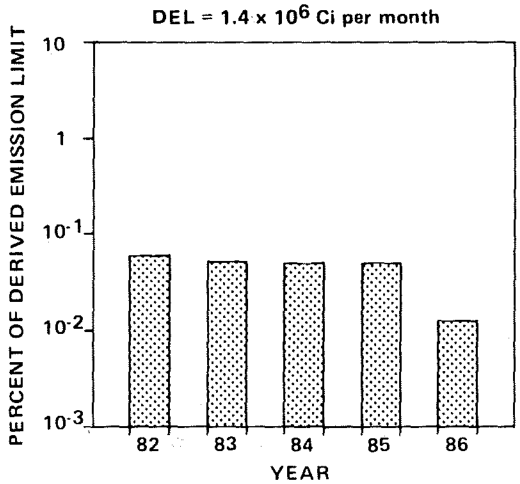
DEL = 5.6 Ci per 7 days



WATER

TRITIUM

DEL = 1.4 x 10^6 Ci per month



GROSS $\beta\gamma$

DEL = 7.5 x 10^1 Ci per month

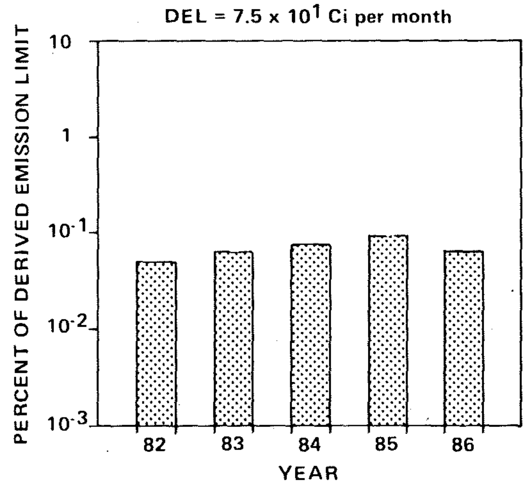


FIGURE 8-6
PICKERING NGS-B RADIOACTIVE EMISSION TREND

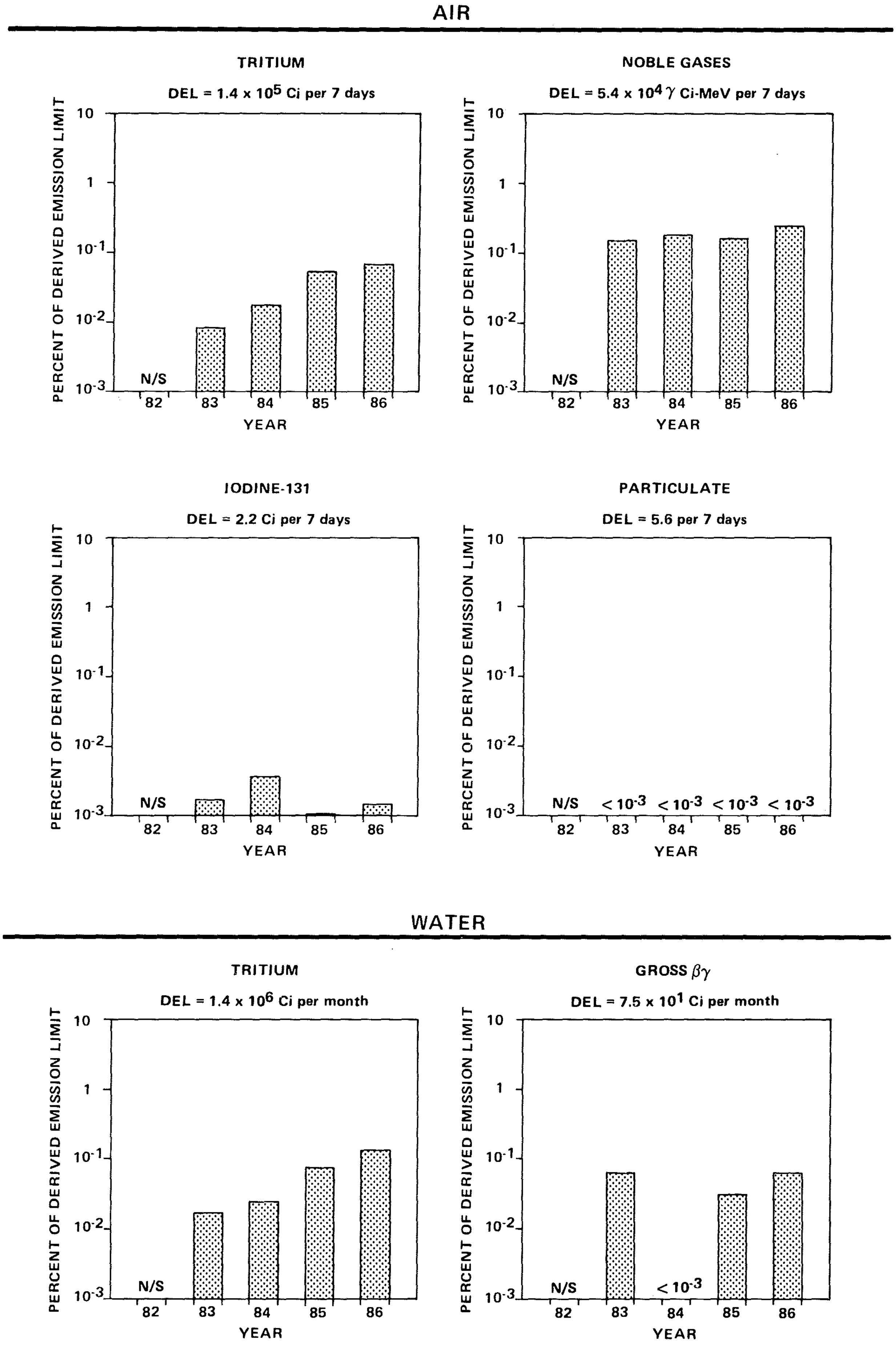
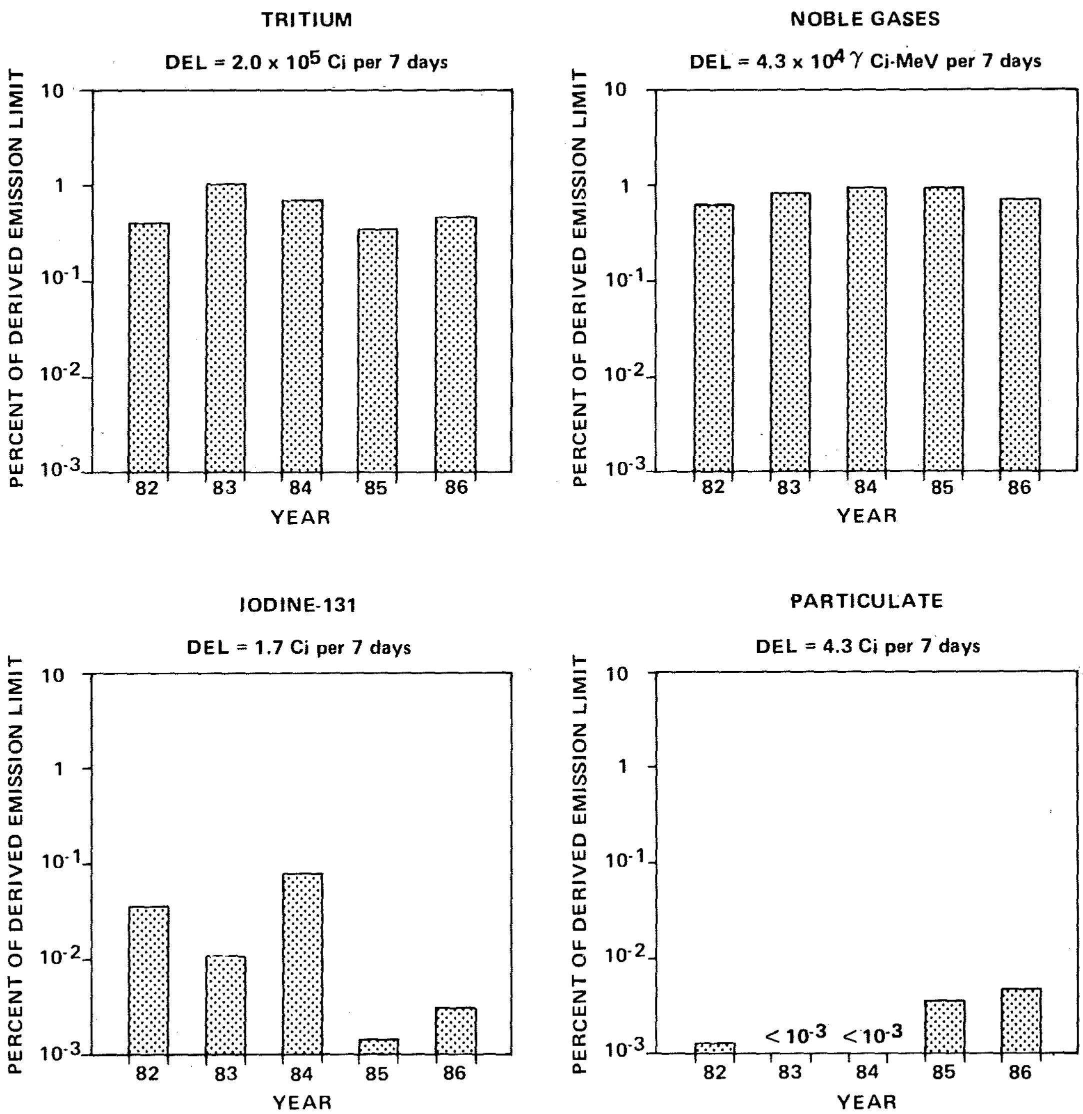


FIGURE 8-7
BRUCE NGS-A RADIOACTIVE EMISSION TREND

AIR



WATER

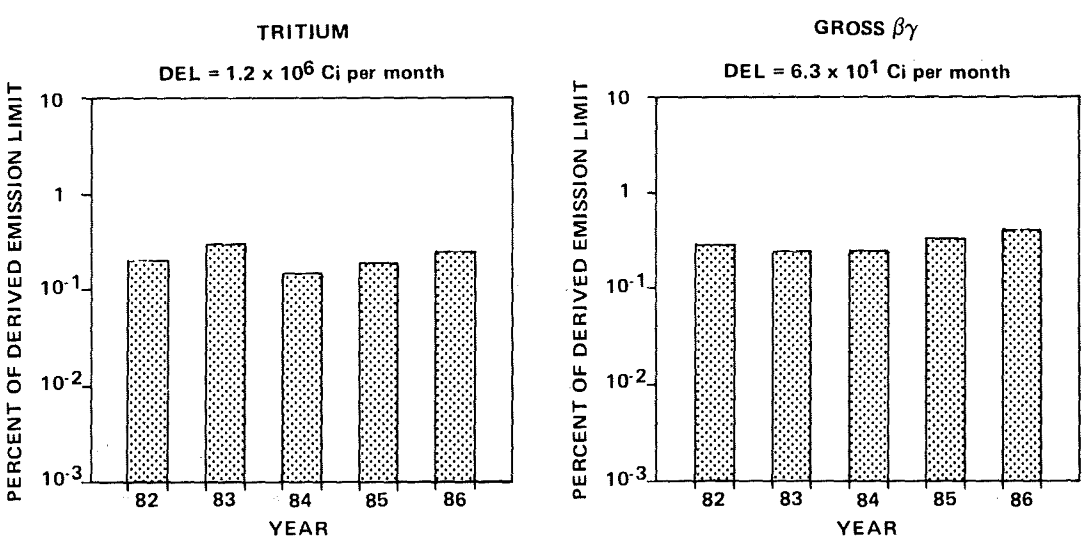
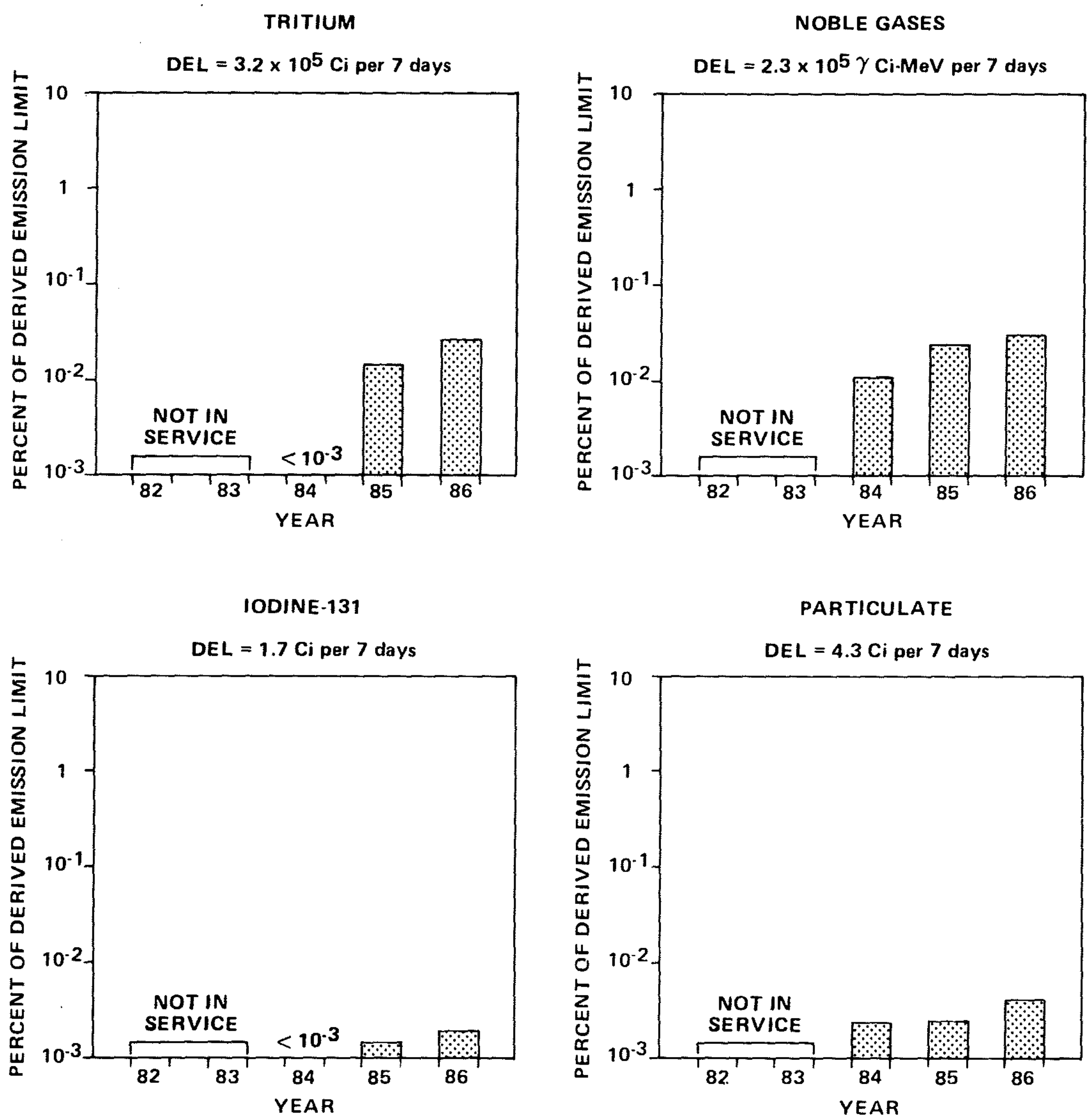
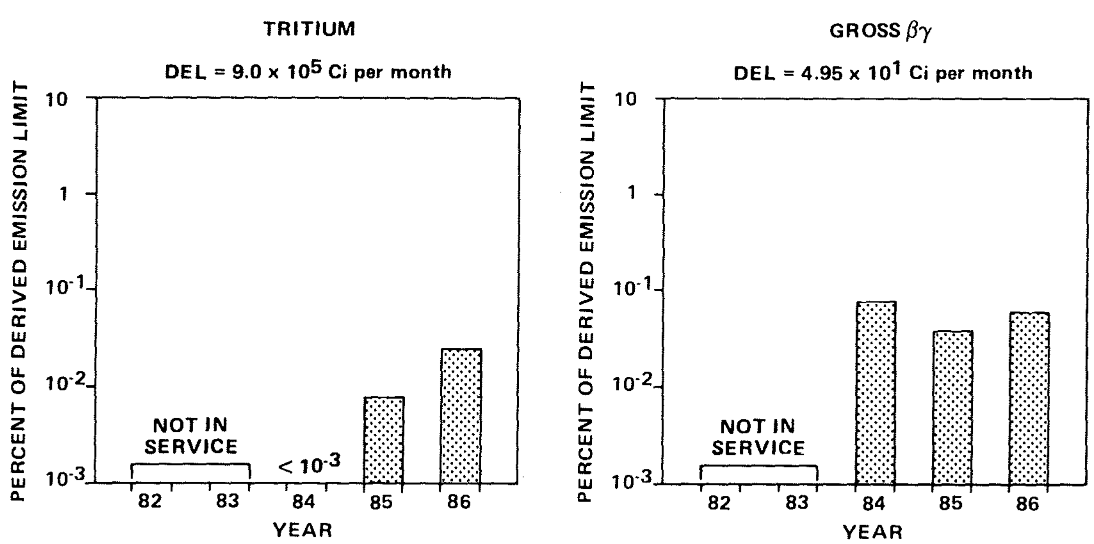


FIGURE 8-8
BRUCE NGS-B RADIOACTIVE EMISSIONS TREND

AIR



WATER



Pressure Tube Performance

9.0 PRESSURE TUBE PERFORMANCE

9.1 Introduction

The CANDU reactor employs a large number of horizontal fuel channel assemblies, each surrounded by heavy water moderator within the calandria vessel. Each assembly is supported at its ends by the calandria/shield tank structure. A typical CANDU fuel channel (Figure 9-1) consists of a Zr-2.5 percent Nb alloy pressure tube of about 10 cm inside diameter which contains a string of twelve or thirteen fuel bundles through which passes the pressurized heavy water coolant. The pressure tube is rolled at each end into the hub of a stainless steel end-fitting, which in turn is connected via couplings to feeder pipes. Each such assembly goes through a passage created by the reactor end-shield lattice tubes and a calandria tube. The assembly is supported in the lattice tube by a sliding journal/bearing arrangement (referred to as end-fitting bearings), and in the calandria tube by spacers (garter springs).

The calandria tube separates the pressure tube from the heavy water moderator, and is rigidly attached to the calandria vessel tube sheet. The annulus between the pressure tube, the calandria tube, and the end shield lattice tubes, is filled with dry CO_2 gas* and sealed by the channel annulus bellows assembly. The latter is attached to the end-fitting hub and the reactor end-face, thus permitting relative axial movement to accommodate thermal expansion, irradiation growth and creep of the pressure tube. The annulus space is connected to the Annulus Gas System, which monitors the moisture content in the circulating gas, and provides warning of leaks from either the fuel channel, calandria tube or lattice tubes.

Design basis accident analysis of CANDU reactors includes the evaluation of the consequence of a postulated fuel channel failure leading to discharge of steam/water into the calandria vessel. In this scenario, it is conservatively assumed that failure of the pressure tube containing the high pressure coolant results in simultaneous failure of the calandria tube.

The following sections discuss several aspects of pressure tube performance, beginning with the safety issues relevant to pressure tube failure. The accident sequences following such failures are described, together with a discussion of the two pressure tube failures that have occurred in CANDU reactors to date. This is followed by an overview of the potential mechanisms giving rise to such failures, and the large research and development programs in place to enhance our knowledge of these mechanisms and their consequences.

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* Pickering NGS Units 3 and 4 use nitrogen gas.

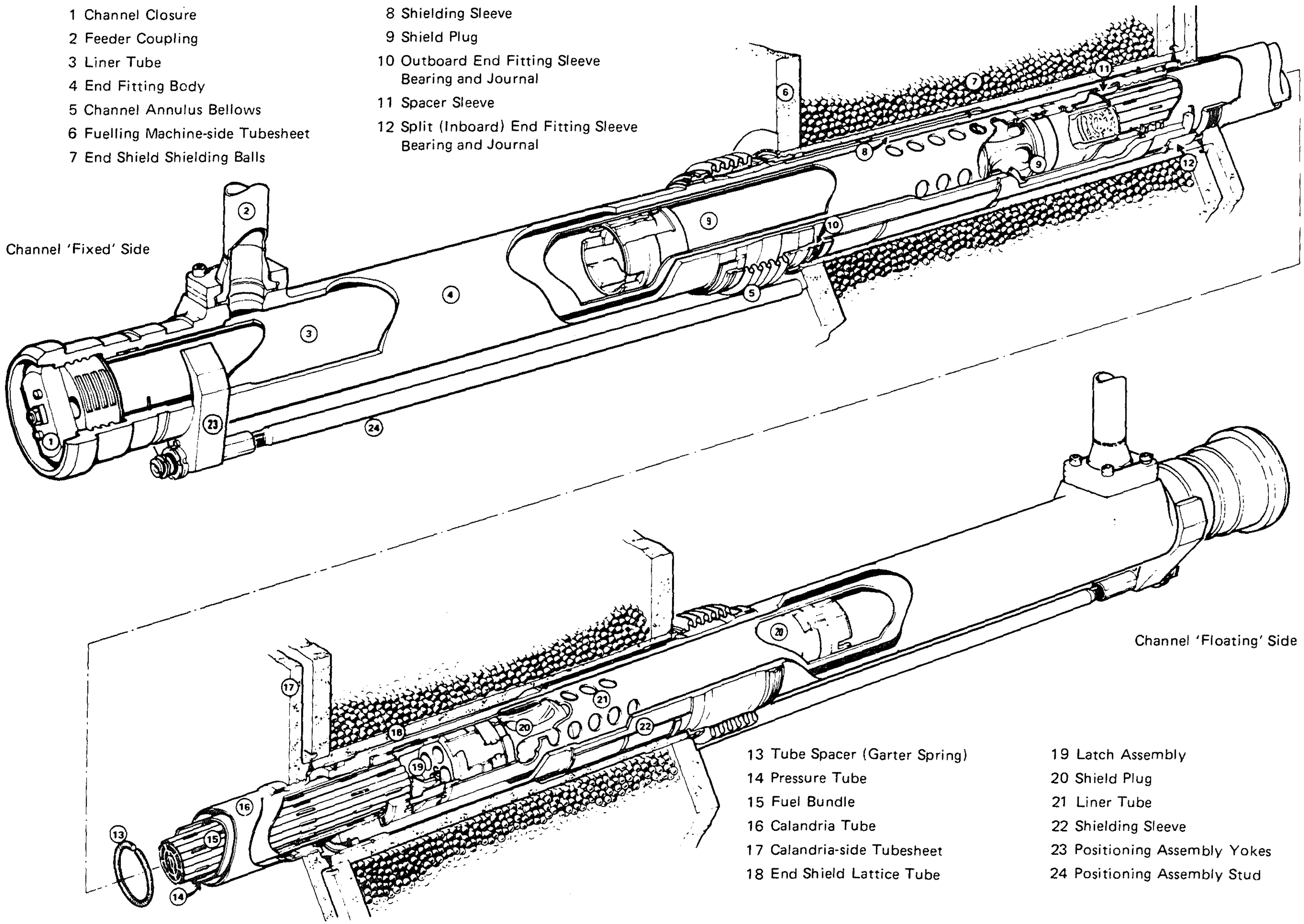

FIGURE 9-1
Fuel Channel Assembly

9.2 Safety Issues and Design Features

9.2.1 Safety Issues of Pressure Tube Failure

Pressure tube failures are treated as a special class of failure in the safety analysis of CANDU reactors. The thermal hydraulic response of the reactor to a pressure tube failure is similar to an equivalent small break LOCA and the system response does not depend on the exact location of the break. However, as the pressure tubes form the pressure boundary within the reactor core, special attention needs to be paid to the consequences in terms of the damage to other in-core structures. As in other accident scenarios, the consequences of this accident are assessed by examining the radiological consequences to the public and by examining the economic consequences in terms of the return to operation of the reactor.

In the case of a pressure tube (PT) failure with the calandria tube (CT) intact, the consequences are dependent on the size of the PT failure. If the failure is small (short through-wall crack) it will be easily detected by leakage into the Annulus Gas System and the reactor will be shutdown as per normal procedure. If the PT failure is large (rupture), it may lead to failure of the channel bellows (as in the G16 failure discussed in Section 9.4.1). The resulting coolant discharge rate is nominally on the order of 20 kg/s and is sufficiently small to allow ample time for operator action to shut down the reactor without invoking any special safety systems. In this type of failure it is to be noted that there are no radiological consequences as the fuel remains well cooled throughout the transient, and fission product release from mechanically damaged fuel (if any) remains within containment. The economic consequences associated with this type of single channel accident are relatively inexpensive as it involves replacement of a single pressure tube which can be accomplished in a relatively short downtime (Pickering NGS Units 1 and 2 excepted for reasons discussed in Section 9.4.1).

In the case of a pressure tube/calandria tube failure, the radiological consequences are minimal in comparison to other stylized accident scenarios considered in the licensing analysis. The economic consequences can be evaluated by assessing the damage to the in-core structures.

The specific safety issues of fuel channel failure are:

(a) Can the failure of one fuel channel lead to a failure of any neighbouring channel?

(b) Can the resulting forces in the calandria vessel cause failure of the calandria vessel itself which might result in the core geometry being disrupted?

(c) Can the resulting forces within the core cause the shutdown system to be impaired to such an extent that it can result in a failure to shut down the reactor or to maintain it subcritical?

These safety issues are addressed by considering the basic modes in which the fuel channel can potentially fail, i.e., fish-mouth rupture and complete guillotine failure of the PT/CT combination. The consequences of these failures, and a brief review of the experimental programs which have investigated this type of accident are discussed in Section 9.3. The inherent design features of the CANDU reactor which tend to mitigate the consequences of such events are discussed below.

9.2.2 Engineered Safety Design Features

The pressure tubes are designed to meet the intent of the ASME code requirements pertaining to nuclear pressure vessels. Based on this design they possess a large strength margin under both operating and upset conditions. The pressure tube material, Zr-2.5% Nb, has a high fracture toughness and can tolerate small cracks without catastrophic failure of the PT.

The G-16 incident clearly demonstrated that the CT can survive a PT failure in normal operation (Section 9.4). Consequently, the CT acts as a primary barrier for containing the PT failure. Failure of the CT would result in pressurization of the calandria vessel. In all CANDU reactors this vessel is equipped with rupture discs which limit the pressure rise due to a fuel channel failure.

Another design feature which limits the amount of core damage is the core layout itself. The close spacing of the pressure tubes and the many incore devices (such as the guide tubes for shut off rods, control absorbers, adjuster rods, etc.) limits the amount of movement of a failed channel. This effectively restricts the potential core damage zone due to pipe whip in case of a guillotine failure of a channel.

The reduction in reactivity worth caused by the inability to insert some shut-off rods fully in damaged guide tubes is shown by analysis not to affect the functional ability of SDS-1. The second shutdown system (SDS-2) is activated by injection of gadolinium into the moderator through several poison injection nozzles. As there is no credible mechanism for blocking each of these nozzles during the event, the ability of SDS-2 to provide an independent shutdown capability is always ensured. Another engineered safety system which adds to the system subcriticality is the Emergency Coolant Injection (ECI) System. The ECI System is activated after reactor trip and this results in addition of light water to the moderator through the break. As the light water is a strong neutron absorber, it will increase reactor subcriticality, and ensure that the reactor stays shutdown in the long term.

9.3 Accident Sequences Considered

In the licensing and safety analysis related to pressure tube or fuel channel failure of CANDU reactors the following two categories of accidents are considered: spontaneous rupture of the fuel channel with the reactor under normal operating conditions and failure caused by very high fuel temperatures resulting from prolonged blockage of the fuel channel or an undetected feeder pipe leak causing flow stagnation in the channel.

The consequences of other accidents, for example those resulting from loss of reactivity control (LORC), are limited by reactor trip parameters designed to prevent, or limit, fuel dryout. Therefore there is a large margin to pressure tube failure for these events and they are not considered further here.

Analytical investigation of pressure tube integrity (Reference 1) for the second category of accidents indicates that the pressure tube is likely to fail before any fuel melting due to development of large circumferential temperature gradients. The potential for pressure tube failure before the fuel attains high temperatures is conservatively ignored in order to maximize the consequential damage. This type of bounding approach (termed limit consequence) yields the worst consequences for any accident scenario. An example of this approach is the flow blockage analysis reported in Reference 2 where the entire fuel inventory of the channel is assumed to reach melting (≈3000°C) before channel failure occurs.

In both accident categories the sequence of events following the fuel channel rupture are very similar. In the early phase of channel failure the discharge of coolant into the cool moderator causes a transient pressure loading on all the surrounding structures. This transient phase of loading is referred to as the hydrodynamic transient. The ejection of fuel can result in impact, or in the case of molten fuel ejection, ablation (surface melting) of the adjacent in-core structures. Subsequent to this jet and thrust forces arise, due to the long-term discharge of coolant through the break, which can interact with other in-core structures by jet impingement or pipe whip. The hydrodynamic transient is dependent on the condition of the fuel at the time of ejection and hence on the assumed sequence of events which lead to channel failure. In the case of a channel failure under normal operating conditions the hydrodynamic loading is relatively mild, whereas in the limit consequence scenario, these loadings are estimated to be relatively severe.

Various experimental programs have investigated the consequences of such fuel channel failures. Analytical modelling of such postulated events is reported in References 3 and 8. A review of the experimental programs and a summary of the analytical investigations is given below.

Historically, pressure tube rupture has been investigated to assess the potential for channel failure propagation, calandria vessel pressurization and damage to in-core structures assuming PT/CT failure occurs. The series of tests conducted in Italy in the Betula test facility (References 9,10) and in the Mark test facility (Reference 11) identified the pressure transient phenomena in the moderator tank. The Betula tests showed that the calandria vessel will be subjected to a very short duration pressure loading. The Mark tests indicated that the shield water reduces the peak pressure loading on the calandria shell. A similar series of tests conducted in Canada (References 12,13,14,15) indicated that channel failure propagation is unlikely, and that damage to the calandria vessel from impact by ejected fuel is minimal. The series of tests reported in Reference 13 used thermite explosions to cause PT rupture, resulting in severe pressure transients within the surrounding water. These conditions can be viewed as a conservative extreme to which energy can be packed into the pressure tube before failure. The results from these tests indicate that even under these severe conditions, channel failure propagation will not occur. A recent series of full-scale tests (References 16,17,18,19) investigated the behaviour of the calandria tube following pressure tube rupture. Both simulated and prototypical Zircaloy calandria tubes were employed in the tests. The results indicate that the calandria tube can survive a pressure tube rupture under reactor operating conditions.

Analytical investigations of the consequences of fuel channel failure (References 2,3,4,5,6,7,8) consider the combined effect of the various damage mechanisms on the surrounding in-core structures and the calandria vessel. For example, to evaluate the adequacy of SDS1, a shut-off rod (SOR) guide tube is considered damaged if it experiences a dent deep enough to hinder the insertion of its rod. The net reactivity worth of the available SOR's is shown to be adequate for any location of break and for all Ontario Hydro reactors.

Based on analysis and experiment, it is concluded that:

(a) Any fuel channel failure will be confined only to the affected channel. Channel failure propagation will not occur.

(b) The calandria vessel will remain intact, hence, there is no possibility of disruption of core geometry.

(c) Shut off rods (SDS1) remaining available after the accident can safely shut down the reactor, and maintain it subcritical, with no credit taken for SDS2.

9.4 Pressure Tube Failures

9.4.1 Pickering Unit 2 G16 Failure

On August 1, 1983, Unit 2 at the Pickering NGS experienced a sudden failure of Zircaloy-2 pressure tube while operating at full-power (Reference 20). The initial indication of a leak came from the automatic activation of the feed pumps that add heavy water to the primary system, and was quickly confirmed by the observation of a steam-filled vault. The operator noted excessive feed to both coolant loops, and observed that the heat transport storage tank level was falling rapidly. The size of the leak, subsequently estimated to be about 18 kg/sec, necessitated the transfer of heavy water from storage at other units. This transfer continued until the recovery pump could be activated to pump the heavy water collecting in the vault sump back into the coolant circuit. The pressure and temperature in the circuit at all times remained within the required limits.

Within 40 minutes of detecting the leak, the operators had successfully reduced reactor power to 2 percent full power and initiated cooldown. The heat transport system pressure had been reduced to a low level to reduce the leak rate and allow pressure control to be maintained. The operators had the unit safely shut down with the reactor on shutdown cooling by 1 hour, 25 minutes after the first alarms. No safety system (reactor shutdown, emergency coolant injection, or active containment) was called for, either automatically or by operator action.

Subsequent reactor inspection revealed heavy water escaping from both ends of the annulus between the pressure tube and calandria tube of channel G16, confirming that although the pressure tube had ruptured, the calandria tube was still intact. Removal of the fuel bundles from G16 required significantly higher than normal fuelling machine ram forces, and subsequent inspection of the discharged fuel revealed that bundles 10 and 11 were each missing a single fuel pencil; this raised the possibility that the bundles had somehow snagged in the damaged pressure tube. The fuel channel was isolated on August 13 by blanking off the feeder piping at each end of the channel. With the leak stopped, the channel was drained, and a miniature television camera was inserted. This revealed an axial crack about 20 mm (0.7 in.) wide at its widest point extending about 2 m (6 ft) from the eastern (outlet) end of the tube and terminating in a 120° circumferential tear. In the crack, which ran along the bottom of the tube were the two missing fuel pencils which were later removed.

The cracked pressure tube was removed and examined at the CRNL (References 21,22,23,24,25,26,27). The crack was located at the bottom of the pressure tube and a number of white, blister-like, features were visible at points near the crack path. These were subsequently identified, using metallography, to be massive precipitates of zirconium hydride.

The conclusions of the overall failure analysis were as follows (References 28,29). The outlet garter spring moved about one metre inboard of its design location, probably during construction or commissioning. This eventually resulted in contact between the pressure tube and calandria tube, due to sag of the former by creep. The pressure tube was cooled at the contact points, causing hydrogen and deuterium to migrate by thermal diffusion and to precipitate there as zirconium hydride/deuteride. This behaviour was aggravated by a very high pickup of deuterium from the heavy-water coolant during operation. This resulted in a high concentration of hydrides near the blisters, which is known to lower the fracture toughness. A crack was initiated by an unspecified mechanism at one or more blisters and grew to an unstable size, possibly by delayed hydride cracking, without penetrating the inner surface of the tube. The thin ductile web on the inside surface later failed, and the crack extended to its final 2 m length.

A program of selective pressure tube inspection and removal for Unit 2 was instituted immediately following the pressure tube failure. On November 14, 1983, Unit 1 (the only other Ontario Hydro reactor with Zircaloy-2 pressure tubes) was shut down for tube checks. A total of 66 tubes were examined with ultrasonic inspection and eddy-current testing for indications of hydride patches and garter spring location, and 12 tubes were removed for laboratory examination. About one-third of the tubes inspected gave strong ultrasonic indications that suggested a high probability of the presence of hydride patches. Of the tubes removed, five had hydride blisters, showed evidence of pressure tube-calandria tube contact, and had displaced garter springs. In view of these results, and the fact that the reactors would need eventual retubing in any case due to axial creep, it was decided in early March 1984 that the quickest way Units 1 and 2 could be brought back into reliable and economic operation was to commence large-scale pressure tube replacement immediately.

The implications of the G16 failure have been the subject of much speculation. However, it is important to separate the design implications from those associated with reactor safety. From the design standpoint, all Ontario Hydro's CANDU reactors built after Pickering Units 1 and 2 use a different material for their pressure tubes, i.e., a zirconium-niobium alloy (Zr-2.5 wt% Nb) rather than Zircaloy-2. For reasons outlined in Section 8.5, reactors built with Zr-2.5 wt% Nb pressure tubes are expected to be less susceptible to a G16 blister type of failure. From the safety standpoint, performance of both reactor and operating crews in response to the G16 failure were good. Design and operating analyses have confirmed that a sudden pressure tube failure in a CANDU reactor, whilst operating at full power, presents no public safety or worker safety concerns. The event was far more benign than assumed in safety and licensing analyses where spontaneous PT/CT failure is postulated.

The G16 calandria tube retained its integrity throughout the incident. Removal and testing of this calandria tube and subsequently another one confirmed that the basic material had increased in strength due to irradiation hardening and had retained adequate ductility. In addition, fixed ended burst testing of the material, showed a combination of irradiation hardening and biaxial strengthening that under pressurization increased the burst strength to about 40 percent above the strength required to resist failure under reactor conditions (Reference 30).

Reactors built after Pickering NGS A employ slightly thinner calandria tubes and generally operate at higher heat transport system pressures. The margin to calandria tube failure, following a G16 type of event, is therefore less than that in Pickering NGS A, but is still in excess of about 20 percent. On this basis, sudden pressure tube failures at full power in any of the Ontario Hydro CANDU reactors are not expected to produce safety consequences different to those experienced at Pickering NGS A, and hence pose no threat to the public.

9.4.2 Bruce Unit 2 N6 Failure

At 11:15 pm, March 26, 1986, the unit operators received an indication of a pressure tube leak in Bruce NGS Unit 2, when heavy water from the heat transport system was detected in the annulus gas system. The reactor was not operating at the time, having shut down automatically at 5:24 that morning due to an unrelated problem. The presence of a leaking pressure tube was confirmed on March 27, 1986 and the unit was cooled down to about 45°C and depressurized. Subsequently on March 28, in an attempt to locate the leaking pressure tube via acoustic monitoring techniques, the unit was repressurized. When the pressure reached about 8.4 MPa, the pressure tube in channel N6 ruptured. The resulting transient, waterhammer-type pressure loading, experienced by the calandria tube caused it to fail as well. This is confirmed by analysis of the pressure tube failure and of the waterhammer loading on the calandria tube (discussed below). The resulting increase in volume of heavy water in the moderator system caused a slight overflow (2 kilograms) which was recovered. There was no risk to station personnel, or the public, at any time during the event.

The fuel bundles were removed from the affected pressure tube and about half the moderator was drained into storage, in preparation for repair, since the damaged channel was half way down the face of the reactor. In-situ video examination of the pressure tube showed that the crack was about 3.8 m long extending from the west outlet rolled joint. A flap of material was seen on the inside surface extending axially about 440 mm from the end of the pressure tube. The flap and crack were at about 11 o'clock, viewed from the west end of the fuel channel. Subsequent inspection of the calandria tube revealed a longitudinal crack along the full length of its seam weld. There was no additional damage to adjacent fuel channels or any in-core structures. The end-fittings, pressure tube and calandria tube were removed from the reactor and sent to CRNL for examination.

The CRNL investigations (References 31,32) confirmed that leakage was caused by the pressure tube being penetrated by delayed hydride cracking (DHC) in the radial direction from a large, shallow angled, lap defect in the rolled joint region. Most of the DHC occurred during the last hot-shutdown of the reactor, resulting in a long crack. When the reactor was cold pressurized, the crack was longer than the critical crack length, and the tube ruptured. The presence of a manufacturing lap defect was also confirmed in the corresponding off-cut from the as-installed pressure tube.

The failure mechanism of the calandria tube is best understood from thermal-hydraulic as well as material behaviour considerations (References 16,17). Following a sudden pressure tube rupture, the pressure tube/calandria tube gap fills with water until the annulus space pressurizes. This takes typically about 0.5 seconds, following which water discharges out of the relatively small constrictions at the end-fitting bearings. Any deceleration of water at the ends will result in a transient overpressurization of the annulus. The peak overpressure is related to the fluid velocity before it is decelerated. The duration of the pressure pulse is given by the time it takes for the compression wave to travel to the reactor headers and return to the annulus as a rarefaction wave. The phenomenon described is generally termed a "waterhammer" transient, and is in principle similar to the hammering observed in household water pipes. The colder the water and higher the pressure, the greater the fluid velocity into the annulus, and hence, the higher the transient overpressurization of the annulus. The peak pressure is attenuated to some extent by circumferential straining of the calandria tube. If the resultant peak pressure exceeds the burst pressure at the relevant temperature and strain rate, or if the accumulated strain exceeds the material ductility, the calandria tube will fail. This is precisely what happened in N6 following sudden failure of the pressure tube under cold pressurized conditions.

The above phenomena have been investigated in several full-scale pressure tube rupture tests performed at Westinghouse under the COG/CANDEV program (References 16,19). Models developed to predict the peak pressures and strains observed have been validated against these experiments and applied to the reactor situation. The results show that rupture of the N6 calandria tube would likely not have occurred had the reactor pressure been less than about 8 MPa.

Other aspects of the N6 event (e.g., channel bellows remaining intact) are also explained by both modelling and experiments. The results of the Westinghouse tests show that the waterhammer loading occurs before the region between the bearings and the bellows becomes filled and pressurized. The bearings effectively seal the annulus during the initial filling and subsequent waterhammer pressure loading stage. This is also confirmed by the results of thermal-hydraulic code simulations of the experiments (Reference 17). The absence of bellows failure in N6 is consistent with a failure of the calandria tube during the waterhammer loading as such a failure would eliminate any possibility of pressurizing and failing the bellows.

Following the N6 event, a large program of rolled joint and off-cut examinations was carried out. The investigation confirmed that the probability of a pressure tube containing lap type manufacturing defects similar to that which caused the N6 failure is extremely small. It is also important to note that even with such defects, leak-before-break is likely to occur, as evidenced by N6. Moreover, the attributes of the N6 failure are similar to those of other cracks experienced in Zr- Nb pressure tubes to date viz, delayed hydride cracking originating in overstressed rolled joint areas (Section 9.5.2).

On the operating side, attention focussed on improved operating procedures in the event of a suspected pressure tube leak. These involve quicker identification of a leaking tube through improvements in the gas annulus system and channel monitoring techniques; strict limits imposed on the leak rate at which action is to be taken, and the maximum time at pressure following initiation of a leak; and implementation of reactor specific pressure/temperature operating envelopes during warmup and cooldown intended to avoid fast fracture of the pressure tube.

The improved procedures and continued in-service inspections are aimed at reducing the probability of pressure tube ruptures during full power operation and/or shutdown conditions. However, as both the N6 and G16 experiences show, the risk to the public from such failures is negligible, and certainly well below the design basis limits for these types of accidents.

9.5 Pressure Tube Failure Mechanisms

9.5.1 Introduction

All zirconium alloy pressure tube failures in Ontario Hydro power reactors have involved delayed hydride cracking (DHC). The effect has been observed in hexagonal close packed metals, zirconium (References 33,34) and titanium (References 35,36); and in body centered cubic metals, niobium (Reference 37) and vanadium (Reference 38). The mechanism of cracking requires hydrogen to diffuse up a stress/thermal gradient and accumulate at a stress concentration where hydrides precipitate. If the stress is high enough, the hydrides crack. This sequence is then repeated and the crack progresses in a series of steps. This mechanism has been confirmed with in-situ experiments using electron microscopes (References 38,39). Cracking can only occur at a given temperature if the terminal solid solubility (TSS) is exceeded and hydrides are precipitated.

Historically, the first components to fail by DHC were experimental fuel elements made from Zr-2.5% Nb alloy (Reference 40). The welds between the end caps and the sheathing failed during storage at room temperature. Hydride platelets were seen at the sharp notch at the junction of the weld upset and the sheath. DHC was attributed to the combination of residual stresses from welding and the sharp notch at the weld upset. Most of the examples of DHC in the nuclear industry have occurred in pressure tubes of CANDU-PHWR reactors and these are summarized in Figure 9-2.

9.5.2 Rolled Joints

The pressure tubes in Units 3 and 4 of the Pickering NGS, made from Zr-2.5% Nb, cracked and leaked in 1974 and 1975 (References 41,42) after about two years of operation. All the cracks initiated at the inside

PRESSURE TUBE FAILURES

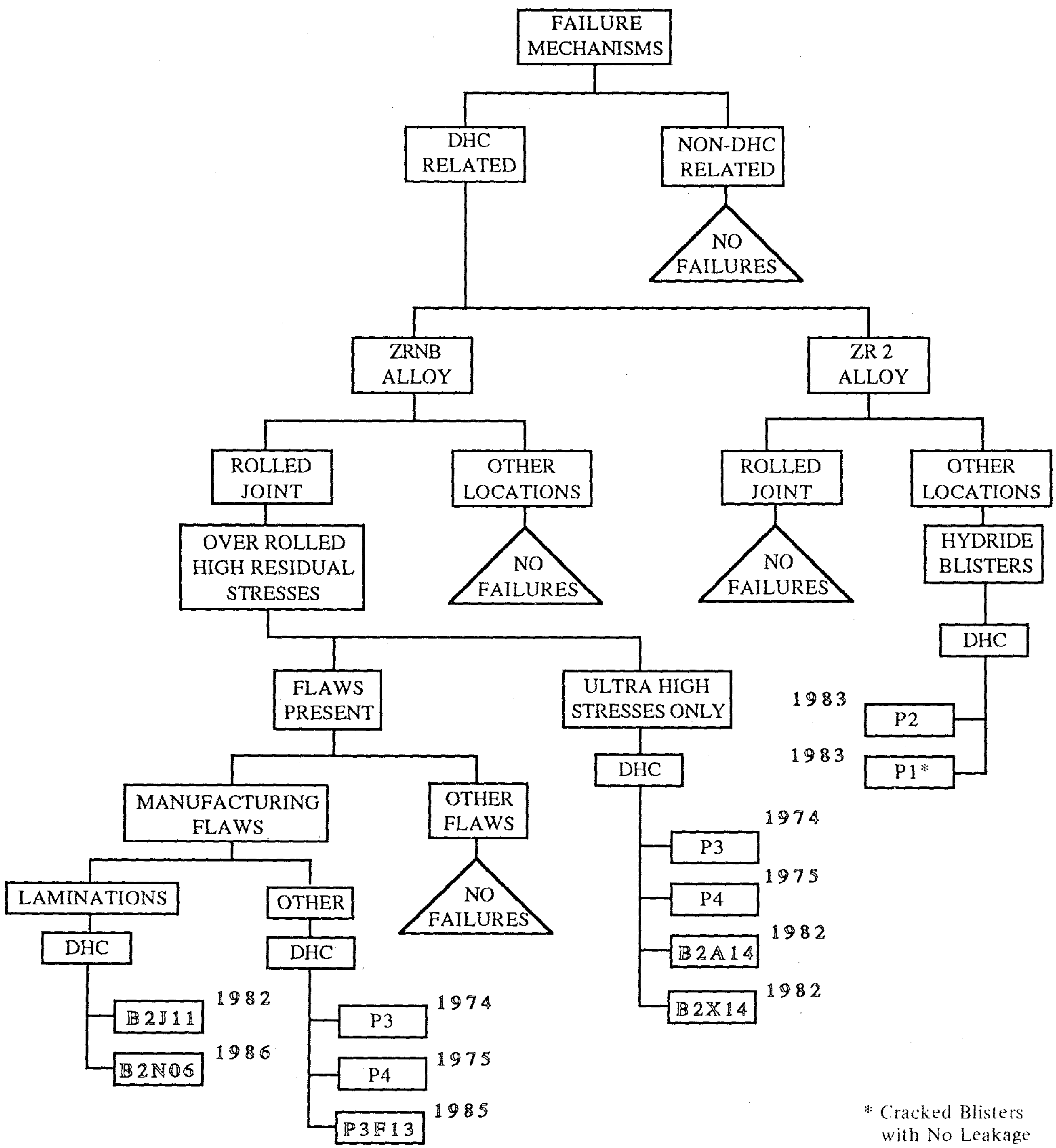

Figure 9-2: Pressure Tube Failures in Ontario Hydro Power Reactors

surface of the pressure tubes where they were attached to the ends of the reactor by mechanically rolled joints. At this position there were high residual tensile hoop stresses, in the range 614-772 MPa in many rolled joints, which resulted from the procedure for making the rolled joint.

The crack surfaces were characterized by coloured, concentric bands, centred about the crack origin, corresponding to portions of the fracture surfaces covered with different thicknesses of oxide. For example, Pickering Unit 3 had three long periods of operation and three major outages or cold shutdowns. The three oxide bands seen on the fracture surfaces were attributed to oxidation which occurred during the three hot operating periods. Cracking could only occur when the pressure tube was cold since all the hydrogen/deuterium was in solution at operating temperatures (i.e., the terminal solid solubility limit (TSS) was not exceeded at operating temperatures).

It was concluded that the cracks had originated early in the life of the reactor, before significant stress relaxation due to hot operation had occurred during operation. Initiation probably occurred after hot conditioning at 250°C, but before criticality, and the cracks grew during the subsequent cold shutdown periods. Seventeen leaking tubes were replaced in Pickering NGS Unit 3 and two leaking and fifty cracked tubes were replaced in Pickering NGS Unit 4, after an extensive inspection of rolled joints.

Although the same rolling procedure was used for Pickering Units 1 and 2, no cracking was observed in these rolled joints up to the time of retubing in 1983. These units however, had Zircaloy 2 pressure tubes. This alloy is also susceptible to delayed hydride cracking, but cannot sustain the high residual stresses necessary to produce cracks because of its lower yield stress.

The rolled joints in the Zr-2.5% Nb pressure tubes of Unit 1 and Unit 2 at Bruce NGS had also been made using the same rolling procedure, but these units had not been put into service at the time the Pickering Unit 3 leaks occurred. To reduce the residual stresses in the rolled joints, these regions were stress relieved. Subsequently, rolling procedures were modified to minimize residual stresses. However, during the 2 years between rolling and stress relieving, the potential existed for cracks to initiate at the inside surfaces of some of the tubes, depending on the level of residual stresses and the existence of stress intensifying flaws.

Three Bruce pressure tubes developed leaks in 1982 (References 43,44,45). The pressure tubes picked up sufficient deuterium in more than five years of service as a result of ingress and corrosion to permit DHC at elevated temperatures (i.e., TSS was exceeded and hydrides were present). The crack surfaces were, therefore, slightly different to those of the Pickering A cracks. During the time between initial leakage and detection of the leak, pressurized water flashed to steam within the cracks producing thermal gradients at the crack tips. This caused the crack to grow at different rates along the crack front, resulting in a crack much longer at the outside surface than the inside.

In 1985, a rolled joint in Pickering Unit 3, channel F13, leaked. The initial residual stresses were estimated at up to 520 MPa, and it is assumed that a small manufacturing flaw, possibly grit from a sand blasting operation, had originally been present at the point of crack initiation (References 46,47).

Examination of the outlet rolled joint from the Bruce N-6 channel (Section 9.4.2) showed that a 77 mm long delayed hydride crack had grown from a large shallow angle planar defect that extended 440 mm from the outlet end of the pressure tube. This defect originated as a small manufacturing flaw (lamination) and grew as a result of corrosion, until delayed hydride cracking occurred at the root of the defect when the defect was about 1 mm deep. The crack then grew radially through the wall to leakage. There were several DHC initiation sites present at the root of the defect (References 31,32,48).

9.5.3 Blistered Pressure Tubes

As discussed in Section 9.4.1, the cause of failure of the G16 Zircaloy-2 pressure tube was related to blister formation resulting from contact with the calandria tube. It is believed that the same basic blister failure mechanism might apply to Zr-2.5% Nb pressure tubes. It is known that garter springs are mislocated in some units operating with Zr-2.5% Nb pressure tubes (Reference 27). However, it is expected that these units will be less susceptible to a blister type of failure mainly because of two factors:

(1) much slower buildup of deuterium due to lower corrosion rates and a smaller percentage pickup of the deuterium released by corrosion; and

(2) the hydrides that form near the blisters are oriented predominantly in the circumferential direction, where their effect on ductility is much less pronounced than when they are oriented in the radial direction, as was the case for blisters in Zircaloy 2.

The examination of selectively chosen Zr-2.5% Nb pressure tubes removed from power reactors, Pickering Units 3 and 4 and Bruce Unit 2, has demonstrated uniformly low levels of deuterium in the body of the tube (References 49,50,51). Furthermore, even though the three pressure tubes examined, P3J09, B2P12 and P4K10, had been in contact with their calandria tubes for several years they showed no blisters and no significant hydride concentrations at cold spots. A continuing program of pressure tube inspection and removal is in place to monitor deuterium pickup and the formation of hydride blisters in contacting pressure tubes.

9.5.4 Conclusions

From all the studies and examination to date, the following conclusions emerge:

(a) Delayed hydride cracking is the only failure mechanism common to all pressure tube failures.

(b) Delayed hydride cracking occurs only when hydrides are present at the appropriate temperature and when either a very high tensile stress, or some lower stress together with a stress intensifying flaw is present.

(c) High residual tensile hoop stresses were present in the rolled joints of some reactors as a result of the rolling procedure used during installation.

(d) Some rolled joint cracks were associated with flaws, but others were simply due to the presence of very high tensile stresses.

(e) Hydride blisters were present in heavily deuterided Zircaloy 2 pressure tubes in contact with calandria tubes as a result of the mislocation of spacers during commissioning. These blisters acted as flaws which initiated delayed hydride cracking in the body of the tube, remote from the rolled joints.

(f) It is expected that Zr-2.5% Nb pressure tubes will be less susceptible to a blister type failure mechanism.

9.6 Pressure Tube R&D Programs

An R&D program to provide a better understanding of fuel channel performance, specifically the pressure tube, has been ongoing for many years. In general much of the work has been undertaken at the AECL Research Laboratories at Chalk River and Whiteshell with additional input to the program from Ontario Hydro Research Division. A review of the world literature, undertaken for the Atomic Energy Control Board in 1986 (Reference 52), concluded that, apart from the expertise in Russia, the world expertise on Zr-Nb alloy performance resided in these laboratories and personnel.

The current major concern is related to the phenomenon of delayed hydride cracking, which has been a major contributor to all the pressure tube failures to date (Section 9.5). This has been exacerbated by the corrosion and subsequent deuterium ingress rate.

Also of significant importance to the economic performance of the CANDU reactor are the phenomena of creep and growth. These deformation characteristics cause axial elongation, diameter increases and sag of pressure and calandria tubes. In some instances, particularly in early reactors (Pickering A and Bruce A) this has resulted in significant outages to monitor and correct for axial elongation to ensure correct operation of the fuel channels.

The work program described in this section is divided into five main areas (References 53,54):

(a) Delayed hydride cracking and fracture
(b) Corrosion and deuterium ingress
(c) In-Reactor deformation
(d) Non-destructive examination
(e) Improved components

In addition, data are gathered through periodic and in-service inspection programs and detailed examinations of pressure tubes removed from the reactors because of failures or defects or as part of the planned pressure tube removal/examination program.

A brief overview of the main areas of the R&D program is provided below. The references in each section refer to descriptions of the programs relevant to that area and identified as work programs (WP) of the CANDU Owners Group Fuel Channel R&D Program.

DHC and Fracture

This aspect of the program is aimed at several areas:

(a) Understanding and modelling the initiation, growth and leakage behaviour of cracks in the rolled joint areas of the fuel channel (References WP 0690, 3214, 3126).

(b) Investigating the effect of irradiation and hydrogen/deuterium content on the critical crack length of the material and the rate at which cracks grow by DHC (References WP 3125, 3133, 3187, 6504, 6505).

(c) Past and current assessments of fracture toughness have been based on burst tests of pressure tube samples. In irradiated material this is an expensive method and new methods of testing small specimens are being developed such that statistical approaches to pressure tube failure can be addressed (References WP 3134, 3305, 6506, 6511).

(d) Investigating the potential for blisters to develop in Zr-Nb alloys (References WP 3410, 3412, 3415).

(e) Past failures of pressure tubes were caused by mechanisms which are now understood. However, new cracks may initiate in the future as the pressure tubes age. The initiation of cracks is an area which is receiving increased attention (References WP 3124, 3126, 3132, 3415, 3510).

Corrosion and Hydrogen Ingress

R&D work into the corrosion performance of Zr-Nb pressure tube material comprises the following:

(a) The effects of the local environment (e.g., temperature, flux and heat transport system chemistry) are being addressed in loops at CRNL as well as 'carrier bundles' which are being exposed in operating reactors (References WP 3119, 3180, 3181, 3503, 6317, 6318, 6508).

(b) Data gained from the pressure tube removals in 1985/1986 from Pickering Units 3 and 4 suggest that impurities in the Zr-Nb alloy may have a significant effect on the ingress rate at the high temperature outlet end of the pressure tube. As a result several programs are now ongoing to address this phenomenon. The specification for the material has already been changed to reflect this knowledge. However the data can also be correlated with time and the pressure tube removal program has been modified to assess this (References WP 3307, 3113, 3111, 3190).

(c) The hypothesis for the high deuterium uptake rate in Zr-2 alloys assumes a critical oxide thickness. This is also being studied for Zr-Nb alloys through the growth of thick oxides in the laboratory, and subsequent insertion, via the carrier bundles, into the reactor to obtain the correct environmental conditions (References WP 3110, 3119, 3306, 3506, 6317, 6318, 6508, 6510).

(d) Along similar lines the effect of boiling on corrosion and deuterium ingress is being addressed. Boiling occurs in approximately the outlet third of many fuel channels in Bruce and (in future) Darlington reactors (References WP 6318, 6508).

In-Reactor Deformation

In general, creep and growth of fuel channels affected the maintenance of the early units whose design did not allow sufficient axial bearing travel to accommodate the lifetime elongation. R&D work in this area is centered around:

(a) Understanding the long term irradiation effects on the material deformation characteristics in fast flux test facilities (References WP 0681, 3515).

(b) Updating of an empirical equation for the prediction of creep and growth of the components, (Reference 55) and understanding the basic mechanisms involved (References WP3158, 3161). R&D into creep and growth has resulted in the development of a modified pressure tube manufacturing process (Reference 56) which will provide a tube with less axial elongation (References WP 3303, 3514).

(c) Understanding the effects of load following on fuel channel performance (References WP 3160).

Non-Destructive Examination (NDE)

The development of new NDE techniques is receiving attention because aspects of pressure tube performance, such as the rate of deuterium ingress and effects of irradiation on fracture toughness, can at present only be addressed by pressure tube removal. Other aspects being investigated are:

(a) Improved inspection of manufacturing flaws (Reference WP 0732)

(b) Improved blister detection, which at present can only be achieved if the blister is cracked (References WP 3141, 3513).

(c) Assessment of defect size is generally based on the length of the defect. In assessing the acceptability of the defect the defect analysis takes into account the depth. Techniques are being developed to measure the depth of defects.

Improved Fuel Channel Components

This part of the program encompasses aspects which can be used to avoid or eliminate problems in operating reactors. A major aspect is the incorporation of hydrogen/deuterium sinks into the pressure tubes to lower the level of hydrogen to below TSS (References WP 0712, 3205).

Concluding Remarks

The ongoing R&D program addresses all of the major current concerns associated with fuel channel performance. As the reactors have aged and deficiencies in the design and/or material have become evident, the R&D program has been revised to reflect the new knowledge, gained through operation of the reactor. Results from the R&D program are, continually fed back into the assessments of fuel channel performance to ensure that the reactors are operated in a safe manner.

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Safety Implications of Chernobyl and TMI

10. SAFETY IMPLICATIONS OF CHERNOBYL AND THREE MILE ISLAND

10.1 OVERVIEW

The accidents at Three Mile Island - Unit 2 reactor in Pennsylvania, USA on March 28, 1979 and Chernobyl - Unit 4 reactor in the Ukraine, USSR on April 26 1986, are the two severe core damage accidents that have occurred in operating power reactors. The impacts of these accidents have been felt throughout the world - perhaps, most keenly in those countries with nuclear power programs.

These accidents occurred in nuclear reactor types that are distinctly different from the CANDU-PHWR (Canada Deuterium Uranium - Pressurized Heavy Water Reactor) nuclear reactors operated by Ontario Hydro. Nevertheless, they were significant events that have resulted in a number of detailed reviews and assessments within Ontario Hydro, to ascertain the safety implications and relevance of lessons learned from these accidents to Ontario Hydro's nuclear generating program.

The factors contributing to the conditions which resulted in severe core damage, and the physical phenomena exhibited during the course of these two accidents, are reviewed here. Both the distinct differences, as well as factors common to the two accidents, are discussed with reference to Ontario Hydro's nuclear safety program. As will be shown subsequently, both accidents served to highlight features of Ontario Hydro's program in existence at the time of these accidents which directly addressed many of the safety concerns raised. There were also a number of specific actions instituted in response to the accidents.

10.1.1 SALIENT FEATURES OF THE ACCIDENTS

10.1.1.1 Chernobyl - Unit 4

The Chernobyl accident was a reactor runaway in which positive reactivity was inserted by an inherent reactivity feedback mechanism at a rate that exceeded the rate of compensating negative reactivity insertion from the reactor control and shutdown system. The resultant super-prompt critical power excursion that developed within a few seconds led to a rapid and large release of energy in the reactor core and its explosive destruction. The disassembly of the reactor core, in turn, led to rapid pressurization of the space surrounding the reactor core and resultant massive failure of the upper reactor structure and portions of the localized containment structures enclosing major components of the main reactor coolant circuit. With the destruction of the structures above the reactor, unimpeded fission product release occurred directly to the environment from the severely damaged and partially dispersed fuel elements. An abbreviated summary of the sequence of events in the Chernobyl accident is given in Table 10-1.

A very detailed accounting of the events leading to the accident, the accident sequence, post-accident mitigation and the consequences is provided in the report prepared by Soviet authorities and presented at an International Atomic Energy Agency (IAEA) sponsored post-accident review meeting held in Vienna in the last week of August, 1986 [1]. Additional information, reviews and analyses of the accident have become available subsequent to this meeting [2,3,4,5,6,7,8].

10.1.1.2 Three Mile Island - Unit 2

The Three Mile Island accident was a loss of coolant accident resulting from the failure of a relief valve on the pressurizer to close . The accident was initiated by a trip of a condensate pump during work to correct a problem in transferring spent resins in the condensate polishing system (a system for removing minerals from the feedwater). This led to tripping of the main feedwater pumps and the turbine. The loss of feedwater was compounded by the automatically initiated auxiliary feedwater system not being able to supply feedwater to the steam generators because isolation valves in the supply lines had been left closed following maintenance two days previously. The resultant total loss of feedwater supply to the steam generators led to the loss of the steam generator heat sink.

This sequence of events would not have resulted in the severe core damage that occurred were it not for inappropriate interpretation of the state of the reactor by the operators and their consequent intervention to reduce the capability of an engineered safety system that had initiated automatically, as designed, following the automatic reactor shutdown. Subsequent operator intervention to turn off the main coolant circulation pumps then established the severity of the resulting core damage which occurred more than two hours after the initiation of the accident. The containment structures did not suffer any damage in this accident, despite the fact that a hydrogen burn occurred in containment at ten hours into the accident. A large amount of activity was released from the damaged fuel to the reactor coolant system and, thence, to the reactor building; however, no direct release of activity occurred from containment. The releases that occurred were from the auxiliary building and were significantly lower than the releases from the core (one order of magnitude lower for the noble gases and over five orders of magnitude lower for radio-iodine). An abbreviated summary of the sequence of events in the Three Mile Island accident is given in Table 10-2.

Many post-accident investigations and reviews were conducted in the United States [9,10,11,12,13] and other countries throughout the world, including Canada [14,15]. In addition, the accident and the physical phenomena involved in severe fuel damage accidents have been, and still are, the subject of extensive analysis and research.

10.1.2 SUMMARY OF ONTARIO HYDRO REVIEWS AND ACTIONS IN RESPONSE TO THESE ACCIDENTS

Subsequent to both the Chernobyl and Three Mile Island accidents, detailed reviews were performed and a number of actions were instituted to assure that nuclear safety concerns they raised were being adequately addressed within the Ontario Hydro program. These reviews and the actions in response to the two accidents are summarized here.

10.1.2.1 Chernobyl

Ontario Hydro participated in the IAEA sponsored post-accident review meeting held in Vienna from August 25th to 29th, 1986, both as part of the official Canadian delegation and as part of the international expert group appointed by the International Safety Advisory Group (INSAG) of the IAEA. Documents obtained at that meeting [1] and, subsequent to the meeting, from other institutions within Canada [3] and outside [2,4], as well as material appearing in the open literature [5,6], have been reviewed in depth. In addition, where appropriate, some analysis has been performed to confirm the understanding of physics of the accident [7].

Ontario Hydro's Nuclear Integrity Review Committee (NIRC), following their examination of the accident and its implications, proposed the following actions in November, 1986.

- Initiate reactor core reactivity studies to assess worst possible core reactivity configurations to provide assurance that no reactor core states, interactions or crosslinks, which could lead to unacceptable overpowers, have been overlooked. Within the same context, re-examine the Pickering NGS A shutdown capability, since it has one shutdown system with two shutdown mechanisms, as opposed to the rest of Ontario Hydro's reactors which have two independent shutdown systems.

- Examine the accessibility of reactor protective circuits to operators in Ontario Hydro's CANDU stations, with reference to the ability to inappropriately inhibit protective functions.

- Undertake a thorough review of safety management within Ontario Hydro, including planning, risk assessment and safety analysis, supervision, operational management procedures and practices, and internal/external safety audits.

- Examine and assess both on-site and off-site emergency response capability with specific reference to fire-fighting, radiation monitoring, protection of on-site personnel, contamination and decontamination control, and the medical response capability within Ontario and Canada to handle a severe radiation exposure accident.

Aside from some work in emergency planning, no new safety research and development activities have been established by Ontario Hydro in response to the Chernobyl accident. This is, in part, a consequence of the fact that that no previously unknown phenomena, or dominant phenomena with high uncertainty in knowledge, have been identified from post-accident reviews and, in part, due to the fact that Ontario Hydro already has existing programs in such areas as, accident analysis and risk assessment, fission product release and transport.

10.1.2.2 Three Mile Island

Following the accident at Three Mile Island - Unit 2 many investigations were conducted by institutions and organizations within the United States of America [9-14]. Ontario Hydro established a Three Mile Island Task Group in 1979 to review information available from the accident, to determine implications for the corporation, and to recommend to the Nuclear Integrity Review Committee (at that time the Nuclear Integrity Review Panel) actions which would lead to necessary and desirable improvements, additions or modifications related to the design and operation of Ontario Hydro's nuclear generating stations [15]. This task group compiled and categorized the numerous recommendations arising from the various international investigations of the accident and documented Ontario Hydro's position on the issues raised in these recommendations [15].

Many of the issues raised were already being addressed at that time by ongoing work; for example, the commitment of station-specific, full-scope simulators for each multi-unit station site and their use in operator training, operational information feedback, operator performance evaluation, consideration of human factors in control room design, safety system status monitoring, realistic accident analyses and enhanced modelling of sheath oxidation and fission product release. Additionally, a number of nuclear safety related work programs were either initiated, or enhanced as a result of the Three Mile Island accident. They included: development of abnormal incidents manuals (an enhancement to existing procedures for handling abnormal incidents); increased emphasis in operator training on maintaining heat sinks; reviews of the adequacy of instrumentation during abnormal incidents; tests of acoustic emission monitoring systems; reviews and studies of monitoring of relief valves, heat transport coolant inventory and hydrogen in containment; studies of post-accident hydrogen behaviour in containment; design of containment hydrogen mitigation systems; and testing of heat transport pump performance in two-phase flow at Ontario Hydro's Pump Test Facility at Research Division.

10.2 KNOWLEDGE OF PHENOMENA AND SAFETY CONSEQUENCES

The Chernobyl and Three Mile Island accidents were distinctly different with regard to the physical phenomena leading to the severe core damage sustained in the course of the events. There were also significant differences in the state of prior knowledge of the phenomena involved. These differences are reflected in the level of nuclear safety research and development activities generated as a direct consequence of the two accidents. This aspect is discussed here in relation to Ontario Hydro's nuclear safety program.

10.2.1 CHERNOBYL

The consensus of international technical experts is that no new physical phenomena were exhibited in the course of the Chernobyl - Unit 4 accident. In particular, the physics associated with reactivity initiated power excursion behaviour has been extensively studied and is well understood. Similarly the behaviour of fuel elements when subjected to large, rapid energy depositions has been extensively researched. In addition, the activity release behaviour during the Chernobyl accident is consistent with the current understanding of accident source terms. In particular the fission product releases during the accident are consistent with available knowledge of uranium dioxide - air oxidation, this being one of the key mechanisms controlling the releases following the disruption of the reactor core. The most speculative aspect of the Chernobyl accident is the detailed process whereby, and the sequence in which fuel channel failures and core disassembly occurred. While these details will, in all likelihood, remain a source of speculation they are largely irrelevant to assuring nuclear safety. What is relevant is the means of handling reactivity initiated transients, namely fast, effective and reliable reactor shutdown systems. The support for the preceding statements is provided by the following brief review of the theoretical and experimental knowledge pertaining to reactivity initiated accidents.

The neutron kinetics associated with prompt critical and super-prompt critical power excursions was established in the earliest days of nuclear fission research and fission reactor development. The independent theoretical developments of Nordheim (1945) [16], Fuchs (1946) [17], and Hansen (1952) [18] established the physical kinetics of power excursions that are limited by prompt reactivity feedback due to the energy released during the excursion. Their mathematical equations provided a means to estimate the total energy release, the maximum power attained, and the burst width at half-maximum power during a super-prompt critical excursion resulting from an idealized step increase in reactivity. The validity of these mathematical models for rapid reactivity insertions with prompt reactivity feedback was established from experimental work on power-bursts performed in pulsed research reactors in the United States, such as TREAT (Transient Reactor Test Facility) and TRIGA. Additional experimental data pertaining specifically to the effect of fuel temperature feedback (the so-called Doppler effect) was obtained from tests performed in the SPERT (Special Power Excursion Reactor Test) program. With the development of digital computers and improvements in numerical methods these simple analytical models have been replaced by computer codes that are capable of computing detailed three-dimensional reactor neutron kinetics. These types of space-time kinetics codes are utilized by Ontario Hydro in performing accident analysis for CANDU generating stations [19,20] and have been verified against data from experimental and power reactors [21], as well as commissioning test data. This 42 year history of accumulated theoretical and experimental knowledge of neutron kinetic behaviour during reactivity initiated transients provides a strong basis for understanding the neutron kinetics of the Chernobyl accident.

In addition to the well-established knowledge of neutron kinetics, there is also well-established knowledge of the effects of large, rapid energy depositions during power excursions on reactor fuel elements. This knowledge is derived from the experimental programs in both the United States of America and Japan to quantify the potential for fuel damage during Reactivity Initiated Accidents (RIA). In the United States the experimental data was obtained from tests performed in the TREAT, SPERT and Power Burst Facility (PBF) test facilities [22,23]. In Japan an extensive experimental program on fuel behaviour under RIA conditions has been conducted in the Nuclear Safety Research Reactor (NSSR) facility, the results of which have been extensively reported in the open literature [24-29]. The results from these experimental programs are generally consistent and demonstrate that the extent of fuel damage increases with the amount of energy deposited during the power excursion. The results also indicated a threshold in failure modes at energy depositions in excess of approximately 380 cal/g. At lower energy depositions fuel failures were relatively benign in that there was essentially negligible conversion of the deposited nuclear energy into mechanical energy. At depositions above 380 cal/g the fuel element failures are characterized by fragmentation and dispersal of the fuel, accompanied by significantly larger mechanical energy conversion. Furthermore, the percentage of the deposited energy converted to mechanical energy generally increased with total energy deposited during the power excursion [29]. With reference to the Chernobyl accident this knowledge is consistent with the estimated large energy depositions that occurred in the fuel during the power excursion and the resultant extreme core damage that occurred [1].

With regard to the processes involved in the destruction of the core it is entirely plausible that there was a "steam explosion" induced by the fragmentation of the fuel. This phenomenon, also referred to in the literature as a vapor explosion or energetic fuel-coolant interaction, involves an explosive release of energy due to rapid, coherent heat transfer from finely dispersed hot metal particles to a surrounding liquid fluid [30,31]. As before, this phenomenon has been the subject of extensive research, both with respect to RIA events [32] and fast reactor hypothetical core disassembly accidents [33].

In addition to this base of scientific knowledge related to reactivity initiated accidents and their potential consequences for fuel and core damage, there have been a number of overpower excursion accidents in research reactors, most notably the NRX accident at Chalk River Nuclear Laboratories on December 12, 1952 and the SL-1 accident at Idaho Falls on January 3, 1961 [34]. The NRX accident was of particular importance to the development of the CANDU power reactors because of its major impact on design principles for safety systems, as discussed in Section 10.3.1.

The Chernobyl accident underscores a long established safety design principle that the most effective means of coping with uncontrolled positive reactivity insertion is to provide rapidly-acting, powerful shutdown capability. Furthermore, it is important to assure that the reactor not be placed in a state which will result in the potential for larger than anticipated changes in reactivity. Although the CANDU reactors do not exhibit the wide variations in reactivity feedbacks with operating power level and fuel burnup that existed in the Chernobyl units, Ontario Hydro considers it prudent to re-examine the potential for events leading to unacceptable overpower transients in its generating stations. This study is aimed at determining configurations of the reactor core, such as absorber rod positions and moderator level, for which there may be either a significant change in potential positive reactivity insertion or reduction in the rate or depth of shutdown capability. This study is being conducted to provide assurance that no significant core configurations have been overlooked in the existing accident analyses and that the shutdown system capability is adequate to limit overpower transients. This study directly addresses the situation that occurred in the Chernobyl - Unit 4 reactor during the test preceding the accident.

10.2.2 Three Mile Island

Unlike Chernobyl, the Three Mile Island accident exhibited physical phenomena, of critical importance in the progression of the accident, for which existing knowledge was uncertain. The phenomena experienced during the accident, such as fuel temperature excursions under severely degraded steam cooling conditions, Zirconium-water oxidation and hydrogen generation at elevated temperatures, pump performance degradation under two-phase conditions, and hydrogen combustion, were all known prior to the accident. However the level of uncertainty in the available knowledge and its importance in realistic accident analysis, were considered to be sufficient reasons to warrant establishment of a large research and development effort in the USA and other countries with light water reactor programs, notably West Germany and France. This was further compounded by the need to develop better operator training and operational procedures to handle slowly developing situations during abnormal incidents, and the general feeling that this was an accident that should never have occurred.

During the course of the accident the hydrogen bubble in the reactor vessel was the most widely reported issue and caused the most public apprehension. This concern arose because of the assumption by some people that the accumulated hydrogen could lead to an explosion in the vessel, its possible rupture and subsequent damage to containment. As was later confirmed, these speculative assumptions were not consistent with available knowledge which supported the fact that there was no free oxygen in the vessel and hence no possibility for an explosive hydrogen-oxygen reaction. However, hydrogen was released from the vessel into containment, where volume concentrations of approximately 8 percent hydrogen developed. At 10 hours into the accident a hydrogen deflagration occurred in containment, causing an 193 kPa(g) (28 psig) pressure spike. These events led to a number of major research programs in the United States into hydrogen behaviour in containment, most notably the Electric Power Research Institute (EPRI) sponsored program [35,36].In addition there was a significant effort related to design of hydrogen mitigation systems. The hydrogen issue was addressed in Ontario Hydro's safety analysis and design [37] and in an experimental program at Atomic Energy of Canada's Whiteshell Nuclear Research Establishment [38].

The extremely low release of iodine to the environment, despite the fact that 30 percent of the core inventory was released from the damaged fuel, was initially surprising, particularly when compared to the US Nuclear Regulatory Commission's legislated fission product release models in use at the time. Subsequent comparison to release fractions from other accidents, such as Windscale, NRX, SL-1 and other research reactors, as well as releases from destructive tests performed in a number of research facilities, indicated that the low iodine release fraction was consistent with the presence of water and the chemical state under which releases occurred [39]. In the period since the Three Mile Island accident, a number of detailed re-assessments of the source term associated with severe fuel damage accidents in light water reactors have been conducted; notably those by the American Nuclear Society [40], the American Physical Society [41] and the US Nuclear Regulatory Commission [42].

The Ontario Hydro Three Mile Island Task Group determined that nuclear safety programs in existence in 1979 were addressing the need for performing realistic accident analysis, improving knowledge of fuel sheath oxidation kinetics and fission gas release at elevated temperatures, radio-iodine transport and chemistry, and containment integrity. The research and development activities initiated specifically in response to this accident were studies of post-accident hydrogen behaviour in the heat transport system and containment, and a program of testing heat transport pump performance in two-phase flow.

10.3 SAFETY SYSTEM DESIGN AND OPERATION

Safety system design was clearly an important contributing factor in the Chernobyl accident, as opposed to being an indirect factor in the Three Mile Island accident. However in both accidents operator interference with safety system functions was a key factor. This is discussed below.

10.3.1 Chernobyl

The shutdown system in the RBMK-1000 reactors suffered from a number of design deficiencies.

Firstly, the design basis for determining the shutdown capability precluded the reactor operating conditions that existed in Unit 4 on April 26, 1986. This was a consequence of the fact that designers, while recognizing the unsafe potential when the reactor is operated below 20 percent full power, relied solely on rational conformance with operating procedures to provide adequate safety. This low power range was designated as a forbidden operating region, but no automatic features were provided to assure safety. Given the short time period over which damaging power excursions can occur, especially with the large positive void reactivity holdup in the RBMK-1000 operating at low power with significant fuel burnup, it is clearly inappropriate to place the burden of assuring safety on the operator.

Secondly, the rate of negative reactivity insertion from the shutdown system was limited by relatively low absorber rod drive velocity (0.4 m/s). When the potential exists for rapid insertion of positive reactivity it is necessary that the shutdown system provide high rates of negative reactivity. The rate of negative reactivity insertion in the case of Chernobyl was clearly and tragically inadequate.

It is worth noting that all water reactor types can experience conditions that will cause rapid positive reactivity insertion. This can occur, for example, in a PWR due to rapid reductions in reactor coolant temperature, or a failure that results in the ejection of a control rod from the pressurized core; in a BWR due to coolant void collapse during the overpressure transient following a turbine trip, or control rod ejection; and in a CANDU due to coolant voiding following a large break loss of coolant accident. That these mechanisms exists is merely a reflection of the fact that changes in reactor conditions that, in one direction, will cause a "safe" change in reactivity will, in the other direction, cause an "unsafe" change in reactivity. This was recognized in Canada, particularly after the NRX accident, and led to the development of shutdown systems with the following features:

- Fast acting
- Large depth (amount) and rate of insertion of negative reactivity
- Independent and physically separate from the reactor control system
- On-line testing to assure availability.

Apart from deficiencies in the shutdown capability, the Chernobyl operators disabled many of the reactor trips in their attempts to stabilize the reactor sufficiently to conduct the turbine-generator test. It would appear that the provisions to disable the reactor trips were readily available to the operators in the control room. This, in conjunction with the operators' lack of understanding of the hazardous conditions in which they had placed the reactor, was a key factor that determined the outcome of the accident. As stated by the Soviets at the meeting in Vienna, the designers made major errors in judgement in assuming that an operator would never place the reactor in a clearly hazardous state - unfortunately, the designers' thoughts were never adequately transmitted to the operators.

10.3.2 Three Mile Island

The design deficiencies in the case of Three Mile Island were not as direct and obvious. In fact, the engineered safety systems initially functioned as designed - the reactor was automatically shutdown by a trip on reactor coolant system high pressure following the loss of feedwater, and the high pressure injection system automatically initiated when the reactor coolant system pressure decreased following the shut down. However, the operators' poor understanding of the the actual state of the plant, coupled with deficiencies in the human factors aspects of the control room design such as poorly located and not readily visible plant parameter indications, led to the operators interfering with the correct functioning of the high pressure injection system.

In the United States the post-accident reviews and assessments led to wide-ranging actions and re-direction of their nuclear program. The Institute of Nuclear Power Operations (INPO) was created to provide a centralized co-ordinating body to assist the extremely de-centralized power utilities operating nuclear generating stations. Increased emphasis was placed on operator training with acceleration of full-scope simulator development; extensive efforts were made to improve the human factors aspects of control room design, including establishment of requirements for safety system parameter displays to assist operators during abnormal transients; the use of probabilistic risk assessment (PRA) techniques was greatly expanded; and development programs were initiated to improve reactor instrumentation, relief valve position indication and the use of acoustic emissions monitoring to detect coolant discharge into containment. As has been discussed in Section 10.1.2, Ontario Hydro's Three Mile Island Task Group established that in many areas the safety concerns raised by this accident were being addressed by programs in existence at the time and, also, recommended a number of actions in response to the accident.

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TABLE 10-1
SUMMARY OF THE SEQUENCE OF EVENTS IN THE CHERNOBYL ACCIDENT

Date/Time	Event	Significance of Event
April 25 1986 1:00	Slow Power Reduction to 50% FP initiated	Unit to be taken to 20-30 % full power (FP) for Test of Generator exciter controls.
13:05	Unit at 50% FP	
14:00	Emergency Cooling System (ECCS) valved out	ECCS remained valved out for rest of event - power held at 50% because of grid demands
23:10	Power reduction resumed	
April 26 1986		
00:28	Power drops to 1% FP 30 MW(th)	Spatial control off,setpoint not reset -Xenon transient develops
1:00	Reactor power at 200 MW	Reactor at low power,small reactivity margin- violation of operating principles and procedures.
1:03 to 1:07	Fourth main coolant pumps started in both loops	Initiates steam drum pressure, drum level and feedwater transients - operators attempt to stabilize conditions
1:21:50 to 1:22:30	Feedwater flow to drum decreased	Reduction in subcooling at core inlet will appear approx. 30 seconds later.
1:23:04	Turbine stop valve closed to initiate test	Two pumps in each loop start running down
1:23:40	Power rising rapidly Manual shutdown button pressed by operator	Reactor proceeding to super-prompt critical condition
1:23:44 (approx.)	Core destruction occurs	Large fission product release, external fires initiated.
5:00	Fires extinguished and paired Unit No. 3 shutdown	
27 April 1986 to 5 May 1986	Boron-carbide,dolomite lead, sand and clay dumped on reactor	Core releases start decreasing then increase due to reduced core cooling
6 May 1986	Nitrogen cooling below core established	Large reduction in releases acute phase of accident terminated.

TABLE 10-2
SUMMARY OF THE SEQUENCE OF EVENTS IN THE TMI-2 ACCIDENT

Date/Time	Event	Significance of Event
28 March 1979	Reactor at 97% Full Power normal operation	
4:00:36	Condensate pump trips	Spent resin transfer operation causes pump trip
4:00:37	Feedwater pumps and turbine trips	
4:00:40	Electromatic relief valve opens on over-pressure	This valve will remain open for nearly 4 hours
4:00:45	Reactor trip on high pressure signal	Shutdown as designed
4:00:51 to 4:01:15	Auxiliary feedwater pumps at pressure	No flow delivered because valves in lines not opened after maintenance two days earlier.
4:01:37	Pressurizer level rising rapidly	Steam generators dry
4:02:37	High pressure injection system (HPIS) initiated	Safety system functions as designed
4:04:30	HPIS throttled	Operator thinks system solid because of pressurizer high level
4:08:55	Feedwater block valves opened	Auxiliary feedwater delivery to steam generators
4;15:25	Reactor Coolant drain tank rupture disc fails	Information not readily available to operator because of poor control room location
5:14:20	One set of coolant pumps stopped	
5:41:22	Second set of coolant stopped	Fuel temperature excursion exacerbated and severe damage assured.
6:15:00	Radiation monitors start going off-scale high	
6:22:00	Pressurizer block valve closed	Discharge from pressurizer relief valve terminated.
13:50:00	Containment pressure surge	Hydrogen burn in containment
20:00:00	Reactor coolant system pressurized and one pump started.	Forced circulation cooling re-established.

Environmental Consequences and Biological Effects Under Severe Accident Conditions

11.0 ENVIRONMENTAL CONSEQUENCES AND BIOLOGICAL EFFECTS UNDER SEVERE ACCIDENT CONDITIONS

11.1 Introduction

A small possibility exists that an accident could occur at an Ontario Hydro nuclear station which may cause emissions of radioactivity greater than those under routine conditions. This could lead to radiation exposures of the public living in the vicinity of such facilities and other environmental effects. The possibility of large radioactive emissions, even in a severe accident, is small because of the safety features built in to the design and operation of these stations. These features are described in other parts of this submission. The biological effects of large exposures to ionizing radiation, without regard for the improbability of an event leading to such a scenario, are briefly discussed in Section 11.2 below, but are described in detail in Appendix C. The environmental consequences of large radiation doses are also addressed in Appendix C. Hypothetical accidents which are used as the basis for the design of reactor safety features, and the assessment of accident consequences, are also described. Finally, ways in which the consequences can be minimized are also outlined.

11.2 Biological Effects of Radiation

The sources of exposure to natural background radiation, amounting to a dose of approximately 200 mRem (2 mSv) per year to an individual are well documented (References 1,2,3). Precisely what risk these small exposures present is subject to debate since little evidence is available to indicate harmful effects at low radiation doses (References 4,5). At large doses, however, the immediate effects are known and studies of animals have identified delayed effects, principally the increased incidences of various types of cancer (see Appendix C). Authorities such as the International Commission on Radiological Protection (ICRP) have taken a cautious view in regarding all radiation exposures, no matter how small, as presenting some risk of health effects proportional to the total exposure. At the natural background radiation level, this assumption implies that the individual risk of a health effect is about 1 in 33,000 per year (Reference 2). The ICRP recommends that the maximum additional exposure to the public due to routine emissions of radioactive materials from a nuclear facility be limited to 1 mSv* per year averaged over several years (5 mSv in any one year). The Atomic Energy Control Board (AECB) in Canada bases its limits on those of the ICRP. (The AECB regulations are not yet fully in line with the most recent ICRP recommendations). Ontario Hydro sets itself targets considerably less than the recommended limits, the maximum additional exposure to the public from routine emissions from the Pickering Generating Station, for example, being less than 0.05 mSv per year. The risks accompanying these low exposures are acceptable in comparison to the benefits.

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* 1 mSv = 100 mRem

There are two types of effects which may be observed following radiation exposures: (a) immediate measurable changes to body tissues which increase with the level of exposure. These are seen only with very large exposures received over short time periods; (b) an increased risk of a long term health effect, such as a cancer. These delayed or chronic effects are thought to apply at all exposure levels.

Appendix C documents in detail the current state of knowledge on the biological effects of exposure to ionizing radiation.

11.3 Pathways to Man

There are many pathways by which radioactivity, if emitted from a nuclear facility, can reach man. Direct pathways include irradiation by radioactivity on the ground or in the air and intake of radioactivity by inhalation. Indirect pathways mostly involve the food chain. The particular pathways which might be affected would depend on the type and amount of radioactivity emitted, and the type of food and manner in which it is grown.

Of all the possible radionuclides that could be emitted in a nuclear accident, cesium (Cs-137) is likely to have the greatest long term impact. This gamma-emitting radionuclide, deposited on the ground, would cause a long term elevation in background dose rate. It is also taken up by plants. Uptake is greatest for shallow-rooted species and for organic soils. Plants tend, however, not to concentrate cesium, (Reference 7). Pathways to man for radionuclides emitted to the environment are illustrated in Figure 11-1. This general scheme applies to many radionuclides (including Cs-137) which could be emitted in an accident. The actual amounts transferred in the different pathways depends on many factors involving the physico-chemical and biological behaviour of the radionuclide, and the nature of the environment. In some cases, certain pathways may be absent or insignificant.

11.3.1 Physical Effects

Following a large airborne emission of radioactivity, the main concern with man-made objects like buildings, vehicles, pavements and sidewalks is that their surfaces could become contaminated, thus giving a long term elevation in background radiation levels. A large radiation dose will not in itself induce radioactivity in materials. Surface contamination could be difficult to remove, especially if it permeates surface cracks in concrete or asphalt. If long-lived radionuclides are involved, such contamination could be longer lasting. Surface contamination is a source of whole body irradiation as well as a potential hazard from skin contamination, ingestion and inhalation of resuspended particles.

Objects, soil and water bodies would not be rendered radioactive simply by being exposed to a passing radioactive cloud. The potential for contamination exists, however, if there is significant deposition of

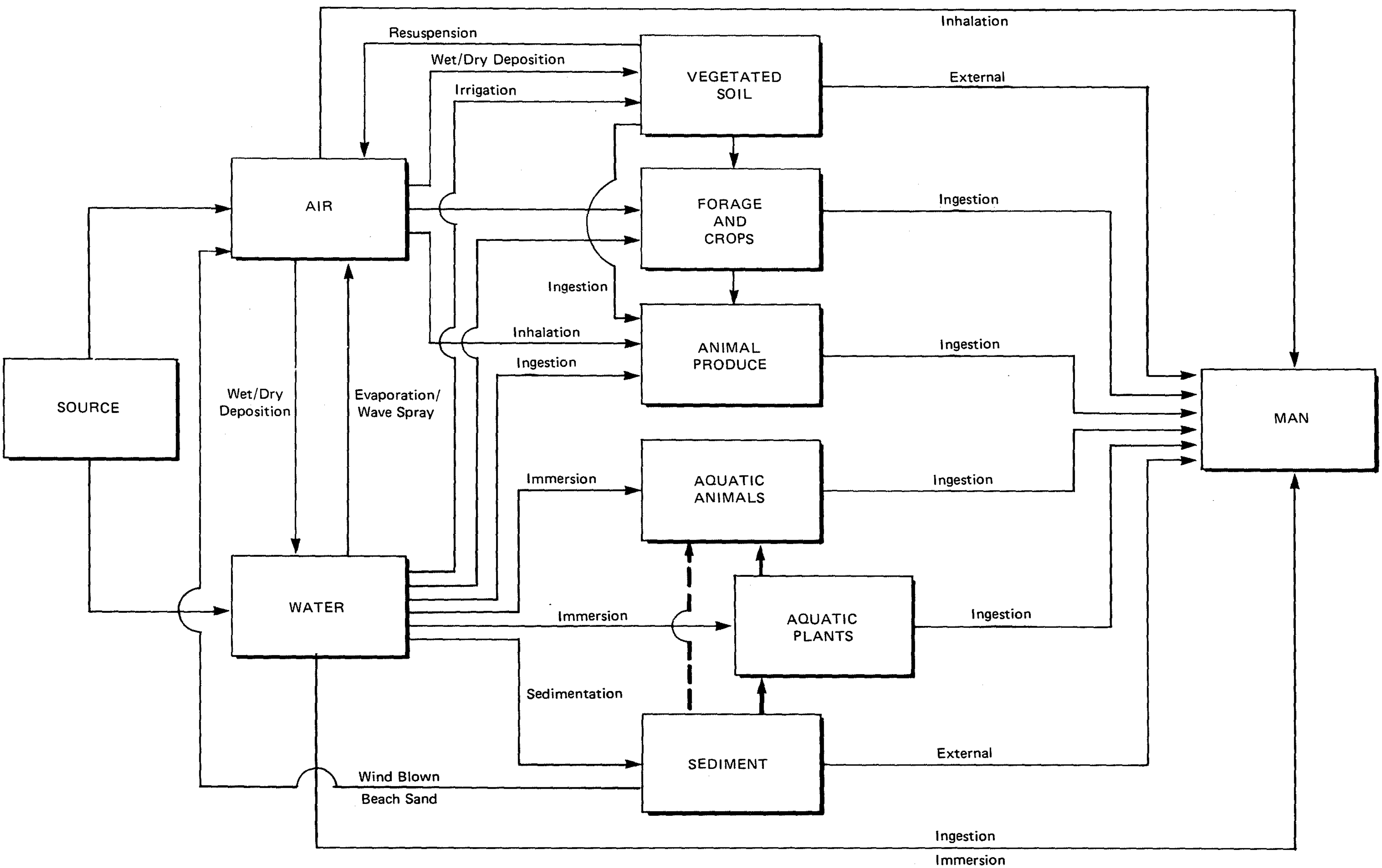

FIGURE 11-1
Pathways to MAN for Radionuclides Emitted to the Environment

radioactive particulate materials or surface adsorption of reactive chemical species. These radioactive substances could then potentially become incorporated in the food chain following uptake by crops, animals or fish. Water soluble radionuclides, deposited on soil and surfaces, would be washed away by rain as surface run-off, or would percolate into ground-water. Eventually such radionuclides would enter streams, rivers and lakes.

For reactor accidents involving liquid emissions, any water-soluble or water suspendable solid radionuclides released would be dispersed and diluted in the lake or river. The concentration at any point in the water body would depend on current and direction, the vertical and horizontal turbulence of the water and the rate of sedimentation in the case of particulates (Reference 7). Considerable removal of soluble radionuclides would take place by adsorption on suspended solids in the water and interaction with silt and other surfaces. For short-lived radionuclides, decay would also be important.

Consideration would need to be given to the possible impacts of the radioactive emission on drinking or irrigation water intakes from the affected river or lake.

11.4 CANDU Design Basis Accidents

The safety design features of Ontario Hydro's nuclear facilities are discussed in Sections 2 and 3.

Briefly, for reactor design purposes, two types of failure are considered: (a) single process system failures; and (b) dual failures where a process system failure is postulated to occur at the same time when a safety system is impaired to such a degree that it cannot perform its function. The single failure accident has relatively minor consequences and will not be further considered here. The AECB has established guidelines for radiation doses for dual failure accidents as given in Tables 1-2 and 1-3, Section 1.

Reactor designs ensure that these guidelines will be met. For reactor licensing purposes, detailed studies of all aspects of station design, reliability and safety are submitted to the AECB in support of application for an operating licence. Exhaustive analyses are performed for various possible failures. The objective is to demonstrate the ability of the design to meet all the applicable criteria assuming the most pessimistic circumstances.

The AECB dose guidelines for the design basis accidents do not specify the detailed nature of the emitted radionuclides (the "source term") or all pathways. This has, however, been the subject of separate studies in which the most likely emissions accompanying various accident sequences have been investigated. These studies and their conclusions are now summarized.

11.4.1 Postulated Airborne Emissions

Accidents involving release from nuclear fuel could lead to high concentrations of gaseous and particulate fission products inside containment. Additionally, large concentrations of tritium could be present from the heavy water coolant and moderator. The relative amounts of radionuclides present in containment would vary widely according to the nature of the accident. Some of the more important of these radionuclides are listed in Table 11-1.

For most credible accident sequences, no emissions would occur for several hours, or possibly many days following the initiating event, since all radioactivity would be in the containment which would be maintained at sub-atmospheric pressure by the vacuum building (References 8,9,10,11).

At longer times, the Emergency Filtered Air Discharge System (EFADS) would be started and pumping would be controlled sufficiently to maintain sub-atmospheric pressure. The exhaust gases would be filtered and monitored before being discharged to the atmosphere. Under these circumstances, radionuclides which are short-lived, soluble in water and which were formed as airborne particulates would not be be a major component of the exhaust gas as these would have deposited on surfaces or been washed out by or dissolved in the large amount of steam and water which would accompany the accident inside containment and by the vacuum building or containment water spray. This is predicted by theoretical studies (Reference 12) and has been observed in accidents such as that at Three Mile Island (Reference 13). A major inquiry into the releases of radionuclides from damaged reactors by a committee of the American Nuclear Society in 1984 (Reference 14), concluded that earlier estimates such as those in the Rasmussen Reactor Safety Study, 1975, (Reference 15), were overestimated by one or two orders of magnitude. More recently, the US Nuclear Regulatory Commission has reassessed the technical basis for estimating source terms (Reference 16), and concluded that reliable computer models now exist with which to predict source terms. For most reactor designs and accident sequences, use of the recommended models give considerably smaller source terms than earlier estimates.

Filtering would further reduce any remaining airborne particulate matter and gaseous radioiodines. The longer-lived of the noble gases listed in Table 11-2 would be the major components of the exhaust gas since these are not very soluble in water and are not removed by surfaces or filters.

Should all emissions occur before the vacuum building operates, some unfiltered gases and noble gases might be emitted. In this case, a potential exists for some of the radionuclides listed in Table 11-1 to be emitted. The probability of significant emissions occurring in this way is, however, quite small since fuel damage required to emit large quantities of airborne radioactivity would not likely occur at once, and the large amounts of steam and water likely to be present would largely remove radionuclides other than noble gases (References 9,12,13).

Table 11-1

Radionuclides Likely to be Present in
Containment Following a Reactor System Accident

Tritium (as tritiated water)

Noble Gases (and daughters):
Kr-85, Kr-85m, Kr-87, Kr-88(Rb-88), Xe-131m, Xe-133, Xe-133m, Xe-135(Cs-135), Xe-135m, Xe-138(Cs-138)

Telluriums:
Te-132

Iodines:
I-131, I-132, I-133, I-134, I-135

Other Fission Products:
Sr-89, Sr-90/Y-90, Zr-95/Nb-95, Ru-103, Ru-106, Cs-134, Cs-137, Ba-140/La-140, Ce-141, Ce-144

Table 11-2

Noble Gases which Could be Emitted
Following a Credible CANDU Reactor Accident

Nuclide (+ daughter)	Half Life(s)	Fission Yield (%)
Kr-85	10.7 y	0.27
Kr-85m	4.5 h	1.3
Kr-87	1.1 h	2.5
Kr-88 (+ Rb-88)	2.8 h (18 m)	3.6
Xe-131m	12 d	0.017
Xe-133	5.3 d	6.7
Xe-133m	2.2 d	0.16
Xe-135 (+ Cs-135)	9.2 h (2,300,000 Y)	6.3
Xe-135m	15 m	1.8
Xe-138 (+ Cs-138)	14 m (32 m)	5.9

Gases or particulates emitted to the atmosphere would be diluted and dispersed such that their concentrations would decrease rapidly with downwind distance. The concentration at any ground level point would depend on such factors as the wind speed and direction, the height above ground of the point of emission and the degree of atmospheric turbulence. Atmospheric turbulence would be determined by the season, the time of day, the amount of cloud cover, and the nature of the terrain (Reference 17). For short-lived radionuclides, decay would also be important. Precipitation (rain or snow) would carry down any particulates and water soluble gases from the radioactive plume or cloud and deposit these on ground surfaces. Some deposition would also occur at a slower rate without precipitation due to gravitational settling and diffusion to ground surfaces, these processes would decrease the plume concentration more rapidly, but could give rise to locally increased levels of ground contamination where precipitation occurred.

11.4.2 Postulated Waterborne Emissions

For accidents involving liquid emissions, any of the radionuclides listed in Table 11-1 could be emitted. Noble gases are, however only very slightly soluble in water and would not be a major component. The noble gas daughters could, however be present in significant amounts. As discussed in 11.3.1 above, these radionuclides may impact local water supplies.

Water supply plant intakes in the vicinity of Ontario Hydro nuclear facilities are located at distances such that even under the fastest water current velocities, several hours would elapse between an emission from a facility to when the leading edge of the aqueous plume reaches the intake. Studies of Lake Ontario off Pickering (References 18,19), indicate that on average, emissions from Pickering NGS take about 20 hours to reach the Ajax water supply plant intake (the closest to Pickering NGS). The least time would be about 2.5 hours in an extremely fast current. Furthermore, having taken in water by the treatment plant, processing further delays by several hours delivery of this water to a member of the public. Processing would also likely remove many of the radionuclides. Thus, even under the worst accident situation, there would be sufficient time to assess the situation and take any appropriate action.

11.4.3 Biological Effects of Severe Design Basis Accidents

Detailed studies for Pickering NGS have been made of the potential emissions resulting from dual failure accidents and likely atmospheric dispersion situations to a probability level of about 10^{-6}. The potential effects of lower probability but larger accidents are discussed in Section 11.5 below. In these assessments, no credit is taken for the effectiveness of any reasonable countermeasures which can be readily implemented.

Following a severe dual failure type accident in a CANDU reactor, exposures to people could be delivered over three distinct time intervals. Firstly, for an accident in which a short term emission occurred (a "puff" emission before the system "boxed-up" and the vacuum building operated), an immediate exposure could occur to those people downwind of the station. Secondly, following a period from several hours to several days, emissions via the Emergency Filtered Air Discharge System (EFADS) would commence. These emissions could be continuous or intermittent, and might continue for several weeks. Thirdly, any radionuclides which deposit on the ground, or are taken up by vegetation as a result of emissions made from the station, could contribute to exposures to people at longer time periods.

For a short term emission at the onset of such an accident, studies have shown that at a distance of 1 km (the station boundary), exposures could be up to about 0.5 mSv whole body dose from noble gas emissions and 2 mSv thyroid dose from radioiodine emissions (Reference 9). In addition, some particulate radionuclides might be emitted, but these would not give rise to a significant immediate exposure (Reference 20). Exposures from noble gases in a short term emission would not be avoidable since there would be no time to take any protective actions. On the other hand, exposures from deposited particulates and possibly iodines could be reduced by appropriate measures taken afterwards. Protective actions and criteria for invoking them are summarized in 11.6 below and are discussed in other parts of this submission (see Section 13).

Operation of the EFADS is somewhat discretionary and could be timed to take advantage of favorable wind directions and atmospheric dispersion. The earliest time a discharge would need to commence is about 1 hour after the accident, the most probable delay being in the range 18 hours to several days (Reference 9). Exposures to people down wind from the accident site would occur as the radioactive plume passes. This would consist of an immediate dose from external irradiation from noble gases, plus the possibility of radioiodine uptake through inhalation, with resultant exposure to the thyroid over the succeeding weeks. Radioiodine could enter the food pathway following deposition on vegetation and appear in milk and other food products. Amounts of radionuclides other than noble gases are, however, not expected to be present in the plume from the filtered air discharge system at significant levels (Reference 20). Again, protective actions could be taken if necessary to minimize or avoid exposures.

The largest doses at 1 km due to the combined effects of a short term air emission and emissions from the Emergency Filtered Air Discharge System (EFADS) are calculated to be about 4 mSv whole body from noble gas, 3 mSv to the thyroid from radioiodine over the period up to one month following an accident (Reference 9), and up to 0.3 mSv in the first year from deposits of Cs-134 and Cs-137 (Reference 20). These values assume no protective actions. Doses would decrease rapidly with distance from the accident site such that by 10 km, they would be less than one tenth the 1 km values under poor atmospheric dispersion conditions.

For radionuclides emitted to water, there is a possibility that human exposures could result from drinking water supplies and from contamination of sport or commercial fisheries. Measures would be taken, however, to monitor and if necessary, close potentially affected water supply plants and restrict fishing. As discussed earlier, adequate time would be available to implement measures before significant exposures could occur.

11.4.4 Environmental Consequences

The effects on the environment from a severe dual failure accident would likely be small. For non-food biota, even under the worst conceivable accident sequences, there is little likelihood of plants or animals being killed over a wide area by exposure to radiation. This is because the levels of radiation exposure would have to be extremely high (see Appendix C for details).

Ground contamination may occur following emission of certain radionuclides. The surface concentrations would be dependent on the meteorological conditions, and in particular should rain be falling during the emission. A study of likely ground contamination following dual failure accidents with average Pickering weather was made using a computer simulation code CRAC2 (Reference 20). The results showed that at 1 km, the total ground contamination would be about 700 000 Bq/m^2, falling to 50 000 Bq/m^2 at 10 km. The major deposited radionuclides immediately after the accident would be Te-132 and its daughter I-132, and I-131. After several weeks, when short lived radionuclides have decayed away, the major radionuclides would be Cs-134 and Cs-137 and the ground contamination would have decreased to about 10 000 Bq/m^2 at 1 km and about 700 Bq/m^2 at 10 km. These values together with the corresponding dose rates are summarized in Table 11-3.

The Cs-134 and Cs-137 which would predominate the surface contamination several weeks following the accident would, in addition to slightly elevating the background dose rate, become mixed into deeper layers of soil and in time enter vegetation, ground water, local streams, rivers and lakes. The quantities involved in such an accident would not, however lead to particularly large concentrations in biota, nor impose significant doses to people via the food chain. For example, the dose to an adult obtaining all his dietary vegetable requirements from land situated at 1 km from the accident can be calculated. Over a period of 2 months to 14 months following the accident, this additional dose would be approximately 2 μSv. This dose is about 0.1 percent of the annual natural background dose.

Table 11-3

Ground Contamination Levels Following a Severe Dual Failure Accident

	At time of Deposition		Two Months after Accident	
Distance (km)	Ground Concentration (Bq/m^2)	Dose Rate** (μSv/h)	Ground Concentration (Bq/m^2)	Dose Rate** (μSv/h)
1	700 000	6.0	10 000	0.035*
10	50 000	0.4	700	0.003*
20	20 000	0.15	250	0.0008*
50	4 000	0.035*	70	0.00003*

** Whole body effective dose in addition to the background of about 0.04 μSv/h from naturally occurring radionuclides in the ground.

* This additional amount is less than the contribution from natural background.

11.5 Larger Accidents

The detailed studies of severe dual failure accidents discussed in this section (References 9, 20) show that the doses to individuals at the boundary fence, assuming they remain there continuously for one month following the accident, are likely to be at most only 20 percent of the AECB dual failure limit of 250 mSv. For any reasonable behaviour consumption, the doses would be much lower.

Thus, realistically, there is a large margin between the highest dose that could reasonably be expected from these analyzed severe accidents and the 250 mSv limit. In order to cause a dose approaching this limit, it follows that a proportionately more severe level of core damage would be necessary, or a smaller but still significant level of core damage accompanied by a containment impairment. Such consequences would be the result of only the most extreme combinations of multiple system failures. These sequences are considered highly unlikely and are not required to be analyzed as part of the station licensing process.

The AECB whole body dose limit of 250 mSv is about one tenth the dose at which individuals may begin to be subject to fatal effects. Thus, an accident many times more severe than the design basis dual failure accidents would be necessary before immediate life-threatening radiation exposures could be possible for any member of the public off-site.

11.6 Counter Measures

In the event of a severe nuclear accident, planned actions (described in further detail in Section 13) can be taken to ensure that doses to members of the public are minimized.

No credible accident to an Ontario Hydro nuclear facility is expected to produce exposures requiring evacuation. However, detailed plans are in place for exposures up to 250 mSv (25 times the level calling for evacuation). At smaller exposures, which are thought to be possible, sheltering inside (with doors, windows and ventilation systems closed to minimize inhalation of airborne radionuclides), thyroid blocking with stable iodine pills and banning of food and water consumption from affected sources may be recommended. These measures are set out in the Province of Ontario Nuclear Emergency Plan (Reference 21).

The actual doses received by a member of the public would likely be considerably less than those given above in Section 11-4, which assume no counter measures.

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18. Ontario Hydro, "Winter Thermal Plume Study, 1978-1979 at Pickering Generating Station, Joint Report, Ontario Ministry of the Environment, Environment Canada - National Water Research Institute, March 1981.

19. Burchett, W.L., "Lake Ontario Nearshore Currents in the Vicinity of Pickering NGS-B, 1982-1984", Ontario Hydro Hydraulic Studies and Development Department, Report No 85211, 1985.

20. Ontario Hydro, "The Assessment of Ground Contamination Arising from Postulated Accidents at Pickering NGS as an Input to Emergency Planning", Ontario Hydro Design and Development Division - Generation, to be issued.

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Economic Consequences and Liability

12.0 ECONOMIC CONSEQUENCES AND LIABILITY

12.1 Sources of Economic Consequence

Any unplanned occurrence at a nuclear generating station which leads to loss of production can be associated with an economic consequence. The consequence may be quite severe if the occurrence results in a prolonged unit shutdown or equipment damage is sustained. Within the spectrum of possible events, those that pose a threat to fuel cooling are of particular concern, because of their potential for very high economic consequence arising from the effects of the release of radioactivity inside the station and possibly a subsequent release to the environment.

An accident that involves a breach of the heat transport system could lead to the requirement to initiate the emergency coolant injection system and possibly containment. If a common safety system is initiated, all units would be required to shut down in an orderly fashion until the safety systems are returned to their poised state. Because of the loss of generating capacity from all connected units, there is potential for economic consequence arising from the need to purchase replacement power. If fuel damage and radioactivity release to containment should occur, restart of the units may be delayed while recovery actions are carried out. The costs associated with the recovery of the station, for example, decontamination and clean-up, together with replacement power, constitute a cost referred to as the "on-site" economic consequence.

For a very wide range of accident severity the major economic impact on the public would be indirect, arising from the on-site economic costs. Even for very severe accidents, the direct impact off-site would be small, provided that containment performs adequately, to the extent that the principal release pathway remains through the filtered air discharge system. Only if severe core damage is associated with severe containment failure, such that the controlled release pathway is bypassed, is there a potential for significant off-site economic consequence. Major sources might be; costs associated with population evacuation, loss of local economic resources, land and property decontamination, etc. As a general category, these sources of economic consequence are referred to as "off-site" consequences.

Past experience has indicated that both the on- and off-site consequences could be significantly affected by factors which are less definable and predictable than those discussed above. Public response, reflected in the behaviour of political, regulatory, legal and advocacy groups, may have an impact over and above purely technical or medical considerations. Estimates of the economic consequences of reactor accidents do not normally include such considerations.

12.2 On-site Economic Consequences

Accidents which result in a breach of the heat transport system and damage to the fuel have the potential to distribute radioactive contamination widely within containment and plant systems. The major contributors to economic consequence can be generalized under four headings:

(a) Decontamination and clean-up;
(b) Replacement/repair of damaged equipment;
(c) Storage and disposal of waste;
(d) Replacement power.

The distribution and magnitude of contamination, and hence the cost of the clean-up, depends on the detailed characteristics of the accident. The structures which comprise containment are large and complex, so decontamination could be a major task, especially if much of the work has to be done remotely. Except for the most severe events, equipment damage is expected to be repairable, although procedures will be inhibited if high radiation fields are present. Clean-up can present major problems in terms of waste products generated. Costs arise not so much from high activity waste but large quantities of relatively low level wastes, such as protective clothing, processed water, resins and filters.

The dominating factor in determining economic consequence is the need to obtain replacement power for the unavailable units. It may be possible to restart the undamaged units before the damaged unit if safety systems can be restored and the damaged unit isolated. Nevertheless, it may be necessary to obtain replacement power for many months or years. Depending on the quantity and duration, and the level of surplus capacity within the system, various alternative sources of power may be utilized. Options such as increased coal-fired generation, purchase from other utilities or even de-mothballing of some units could be considered. Current replacement power costs are of the order of $1 million per unit per week, for a Pickering unit, assuming coal as the alternative. For the loss of a 4-unit station in the future, these costs may be significantly larger, because of predicted changes in relative power generation costs and surplus capacity.

The Darlington Probabilistic Safety Evaluation (refer to Section 4.3) contains estimates of economic risk for Darlington based on on-site costs. A representative station recovery network diagram (Figure 12-1) was developed to identify the major tasks necessary to station recovery. The network was evaluated for cost and duration for each of the fuel damage categories used in the DPSE (Table 12-1). Integrated capital costs and recovery duration were estimated (Table 12-2) and replacement power costs accounted for (Table 12-3).

It was found that, even though a severe accident could result in the loss of the station's generating capability for several years, and generate costs of several billion dollars (depending on assumptions concerning future replacement energy costs), economic risks tended to be dominated by the more 'likely' but less severe events. The economic risk from accidents is predicted to be relatively small, of the order of a few million dollars a year per unit. Because there has been no accident of this nature at Ontario Hydro's nuclear stations, these predictions remain speculative. Operating performance better than that assumed in the theoretical studies would lower the predicted risk significantly.

There is some experience of the on-site costs associated with a nuclear accident, notably that arising from the TMI-II event. The budgeted cost over about a 10-year rehabilitation period is of the order of a billion dollars, without taking account for replacement power. This is larger than the DPSE methodology would predict but includes significant R&D costs, costs due to delays beyond the control of the utility, and about $40 million a year for maintenance of the unit in a stable condition.

Although theoretical studies and experience may differ in detail, the order of magnitude of estimated costs is the same. The general conclusion is that on-site economic risks are dominated by relatively high frequency/low consequence events which do not result in significant release of radioactivity from reactor fuel. The predicted frequency of accidents declines more rapidly than the predicted consequence as their severity increases.

12.3 Off-site Economic Consequences

Theoretical studies and experience at TMI indicate that the occurrence of an accident, severe in terms of fuel damage, will result in essentially negligible release to the environment if containment remains intact (Reference 1). Although it is possible that some off-site costs might arise, for example due to precautionary evacuation, these are likely to be small relative to on-site costs.

Prerequisites for significant off-site costs are a large release into containment accompanied by, or followed by, a major containment failure. For a multi-unit CANDU containment, a failure which results in the predominant leakage pathway by-passing the filter discharge system could result in significant off-site contamination. Because of the high degree of independence between containment systems and the rest of the plant, the predicted frequency of such a release is very low (typically less than 10^{-6} per reactor-year).

Given such a release to the environment, the potential exists for off-site economic consequences to become significant. No detailed studies have been carried out in Canada but, for general discussion purposes, the results of U.S. studies (e.g. Reference 2) can be used. Typical contributions included in the evaluation of off-site economic consequence are; population evacuation and long-term relocation, agricultural product disposal, decontamination of land and property, land area interdiction and health effects costs.

The results of the studies indicate a consequence of up to about a billion dollars for a semi-rural site. Sensitivity studies indicate that this might increase by up to a factor of 10 for a site located near to major population centres. The largest contributor is the cost of land and property decontamination (50%). However, from an overall economic risk point of view, the off-site component appears to pose a lower risk than the on-site component.

The general conclusions to be drawn from economic risk studies are:

(a) Unlike public health risks, economic risk appears to be dominated by relatively high frequency/low consequence events.

(b) The dominant contributor is the "on-site" component. At Ontario Hydro, the cost of replacement power is predicted to be the largest potential contributor to the on-site consequence.

12.4 Liability

Under the provisions of the Nuclear Liability Act of 1976, Ontario Hydro, as owner and operator of nuclear power generating stations, is required to maintain nuclear liability insurance with limits of at least $75 million on each of its operating nuclear stations. This coverage is intended to meet claims by citizens in the event of a nuclear incident at a nuclear facility. It does not cover on-site costs. Ontario Hydro chooses to self- insure with respect to on-site property damage, replacement power is not normally insurable.

The Atomic Energy Control Board (AECB) is responsible for recommending the level of liability coverage to the Treasury Board. The Nuclear Insurance Association of Canada (NIAC) is the sole source from which an operator may purchase liability insurance. The NIAC is a consortium of several insurance companies licensed to do business in Canada. The cost of annual premiums, resulting in $75 million of protection, depends on population density and property values near each station. The cost of annual premiums is reflected in Hydro's rates. For 1986 the premiums totalled $1.7 million.

In the event of a nuclear incident where claims are expected to approach or exceed $75 million, the Nuclear Liability Act specifies that a Nuclear Damage Claims Commission be established to settle all claims arising from the incident. If claims exceed $75 million, the Minister involved is authorized to make payment out of the Consolidated Revenue Fund.

When the Nuclear Liability Act was proclaimed, the Minister directed that a review be made after five years to see if changes were necessary. A review was initiated in 1982 and the AECB released a consultative document on the Act in 1984. The comments received back on this document have been reviewed by an interdepartmental working group, which is working on a list of recommendations for the AECB. Because of a recent legal challenge of the Act, however, there is unlikely to be any further activity on the review until after the court challenge is completed.

Ontario Hydro is reviewing its internal policy towards liability insurance. The view is currently taken that the most cost-effective approach would be to self-insure for the coverage provided by the Act or any increase intended to accommodate change in monetary values since 1976. This would require a change in the Act.

Canada is not a party to either of the two international conventions governing international nuclear liability. The IAEA's standing committee on civil liability for nuclear damage is attempting to draft a protocol governing transboundary liability. However, Canada has declared the U.S. to be "reciprocating country" under the Nuclear Liability Act, thereby extending the same rights to affected citizens whether they be U.S. or Canadian. Canadians may also claim under the American Price-Anderson Act.

<u>References to Section 12</u>

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Table 12-1

Fuel Damage Category Descriptions

Fuel Damage Category #	Release Magnitude(1) BE(2)	PM(3)	Description
1	15	30	Large LOCA, total and indefinite loss of ECI injection.
2	5	10	As above but for small break or equivalent.
3	2	3	Fuel cooling lost at least 1 hour after reactor shutdown. Typical event is failure of emergency coolant recirculation.
4	2	4	Large LOCA with near-instantaneous release due to early flow stagnation.
5	1	2	Large LOCA but with potential for delayed release after steam blowdown.
6	.2	1	Single channel event. End fitting failure with fuel ejection to reactor vault.
7	.001	1	Single channel event. In-core break with fuel ejection to moderator.
8	.0001	.4	Loss of cooling to irradiated fuel in the fuelling machine.
9	0	0	Small LOCA without significant additional fuel failures.

Notes

(1) Approximate release of long-lived, alkali metals expressed as a percentage of nominal, equilibrium core inventory.

(2) Best-estimate

(3) Probable-maximum

TABLE 12-2

Preliminary Numerical Estimates of Capital Cost and Outage Duration

Fuel Damage Category	Unadjusted Capital Cost(1) (M$) BE(2)	(PM)(3)	Outage Duration (Months) 4 - Unit BE	(PM)	Additional 1 - Unit BE	(PM)
1	160	(250)	45	(72)	65	(126)
2	160	(250)	37	(61)	64	(127)
3	90	(140)	24	(40)	21	(39)
4	90	(140)	24	(43)	21	(39)
5	80	(120)	21	(38)	9	(17)
6	10	(20)	17	(31)	0	(0)
7	10	(20)	6.5	(19.5)	3	(7)
8	1	(2)	1.5	(7.5)	0	(0)
9	6	(10)	4	(7.5)	1	(1.5)

Notes:

(1) Unadjusted sum of capital cost components task evaluation
(2) Best-estimate
(3) Probable-maximum

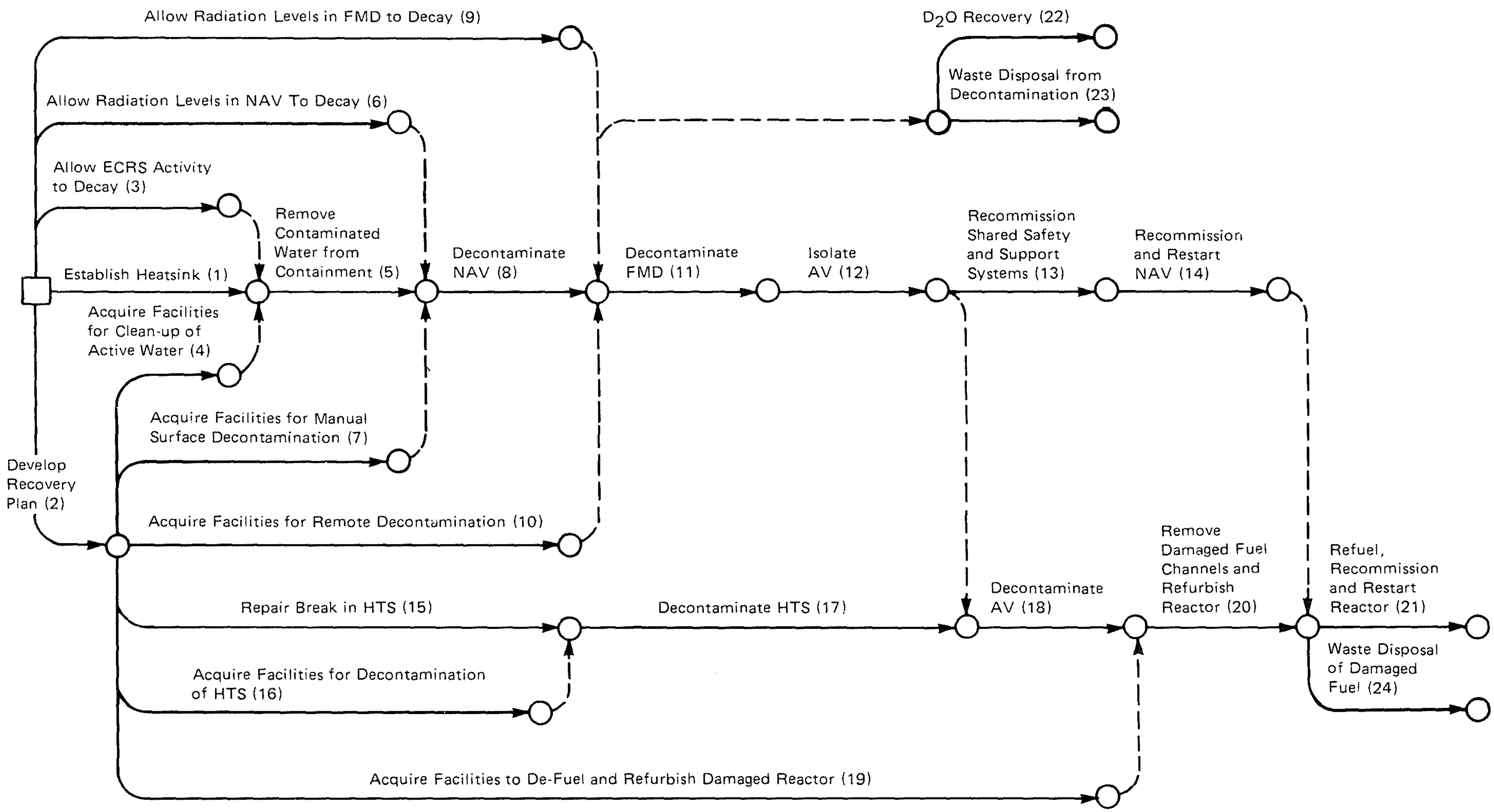
TABLE 12-3

Preliminary Cost Estimates

Fuel Damage Category	Adjusted Capital Cost (1) (M$) BE (2)	(PM) (3)	Replacement Power Cost (M$) 4 - Unit BE	(PM)	Additional 1 - Unit BE	(PM)	Total Cost BE	(PM)
1	260	(690)	2700	(4600)	780	(1500)	3700	(6800)
2	260	(690)	2400	(3800)	770	(1510)	3400	(6000)
3	150	(390)	1600	(2600)	250	(470)	2000	(3500)
4	150	(390)	1600	(2800)	250	(470)	2000	(3700)
5	130	(330)	1350	(2500)	110	(200)	1600	(3000)
6	20	(55)	1100	(2100)	0	(0)	1100	(2200)
7	20	(55)	400	(1300)	40	(80)	460	(1400)
8	2	(5)	72	(350)	0	(0)	75	(355)
9	10	(25)	200	(350)	12	(18)	220	(390)

Notes:

(1) Adjusted by correction factors for overheads
(2) Best-estimate
(3) Probable-maximum



Notes:

1. Numbers in brackets are Task Identification Numbers and Subsection Reference Numbers (5.2.x).
2. AV = Accident Vault; FMD = Fuelling Machine Duct; NAV = All Other Containment Structures

FIGURE 12-1
Post-Accident Recovery Network Diagram

Emergency Preparedness and International Impacts

13.0 EMERGENCY PREPAREDNESS AND INTERNATIONAL IMPACTS

13.1 Introduction

Emergency preparedness has been an integral part of the operation of Ontario Hydro nuclear stations since the initial preparations for NPD-NGS.

The organizations and requirements have changed during the past twenty-five years; however, the principles underlying provision of an effective emergency response remain the same. The response capability must be able to protect station employees, regain control of the affected systems and mitigate the consequences to the public.

These objectives are accomplished through the effective use of procedures and equipment by trained personnel. These procedures include the following elements: emergency identification and categorization, notification process, assessment of consequences (reactor systems, in-station conditions, and off-site conditions), personnel accounting, and protective action decisions and implementation (both on-site and off-site).

A new Provincial Nuclear Emergency Plan (Reference 1) has been recently approved and issued. Although the plan did not change the principles for emergency response, it did make significant changes to the responsibilities of organizations, the methodology for assessment of consequences, and the training requirements. Ontario Hydro has revised its procedures to comply with this plan. A program to upgrade the procedures to improve their effectiveness and useability is underway.

13.2 Emergency Response Capability

The Province of Ontario, the municipalities adjacent to each nuclear station, and Ontario Hydro have a joint responsibility to ensure that an adequate response capability exists to protect the public in the event of a nuclear emergency. The detailed responsibilities are identified in the Province of Ontario Nuclear Emergency Plan, Part I (Reference 1).

The Province of Ontario, through the Ministry of the Solicitor-General, has the primary responsibility to develop and maintain off-site emergency plans, as well as to ensure an adequate state of preparedness. This is accomplished through the development of implementation procedures for the identified ministries, municipalities, and facilities, the acquisition of the required equipment to support the plans, the training of emergency response personnel in each response organization and the periodic review of the response capability.

The municipalities have the responsibility for preparing and maintaining emergency plans and implementing procedures. The municipalities also participate in the training programs and emergency exercises.

Ontario Hydro is responsible for the development and maintenance of a response organization, implementation procedures, facilities and equipment required to support the response capability, provision of assistance to the province and municipalities in their planning and preparedness, implementation of a training program for Ontario Hydro and external emergency personnel, implementation of a public education program, and participation in Provincial nuclear emergency exercises.

13.2.1 Basis for Emergency Preparedness

Provision of an effective emergency response requires detailed planning to allow trained personnel to use the necessary equipment and procedures to protect the public. The specific requirements for detailed planning and arrangements depend upon the basis for emergency preparedness. The most common approach in establishing such a basis is to estimate the consequences of postulated accidents and then introduce an emergency preparedness safety factor.

Ontario Hydro evaluated the consequences of postulated accidents in a report entitled "The Assessment of Radiation Dose to the Public Arising from Postulated Accidents at Pickering NGS as an Input to Emergency Planning" (Reference 2). This study selected events with an estimated frequency of more than 10^{-6} per reactor-year, and with a potential atmospheric release resulting in an exposure to a critical individual at site boundary of greater than 100 mrem within 18 hours of the accident.

The events which met these criteria were evaluated further. Scenarios considered were a large-break LOCA with worst weather, an end-fitting failure of a high power channel with worst weather, an end-fitting failure of an average channel with impaired containment and average weather, a small LOCA with failure of ECI injection and worst weather, a fueling machine accident with impaired containment and average weather, and an irradiated fuel bay accident with failure to isolate bay atmosphere and worst weather.

Of these events, the end-fitting failure of an average channel with impaired containment results in the largest predicted whole body dose to a critical individual at site boundary over both the initial four hour (0.1 rem) and eighteen hour (0.15 rem) periods.

The Province of Ontario Nuclear Emergency Plan - Part I states that the basis for detailed planning is an accident which would result in whole body dose of 25 rem at 1 km. This represents a significant safety factor for emergency preparedness relative to the technical assessment of the consequences of postulated accidents within the 10^{-6} per reactor year criterion.

This basis established the requirements for emergency planning. Such requirements include the need for detailed exposure control plans within a 10 km Primary Zone, detailed ingestion control plans within a 50 km Secondary Zone, method to alert and inform the public of the need for protective actions, as well as all implementation procedures for such protective actions.

13.2.2 Ontario Hydro Organization

To fulfill its mandate in emergency preparedness, Ontario Hydro established a policy committee, an implementation committee, and a Corporate emergency preparedness group.

The Nuclear Emergency Preparedness Committee (NEPC) sets policy and direction on matters related to emergency preparedness. Its membership comprises the Executive Vice-President - Operations (Chairman), Vice-President - Production, Vice-President - Design and Construction, Vice-President - Corporate Relations, and Manager - Radioactivity Management and Environmental Protection (RMEP) Department (Secretary).

The NEPC also has resource staff which include the Director of Technical and Training Services Division, Director of Nuclear Generation Division, Director of Design and Development Division - Generation; Director of Public Relations Division, and Manager of Safety Services Department, Health and Safety Division. This committee ensures that the appropriate Corporate commitment and resources are in place for Ontario Hydro to meet its responsibilities.

The Nuclear Emergency Preparedness Coordinating Committee (NEPCC) makes recommendations to the NEPC as well as coordinates and monitors the emergency planning activities within Ontario Hydro. The NEPCC also provides a means of ensuring a regular interchange of information among units of the Corporation concerned with matters related to emergency planning. In this way, all necessary activities are performed in an effective and efficient manner. Membership of the NEPCC consists of the Manager of RMEP Department (Chairman), Manager of Operational Reactor Safety - RMEP Department, Nuclear Safety Engineer - Nuclear Studies and Safety Department, Health Physics Services Manager, Manager of Public Communications, and Technical Superintendent of Emergency Preparedness Section - RMEP Department (Secretary).

Both committees meet at least four times per year.

The Emergency Preparedness Section, RMEP Department, is responsible for coordinating Ontario Hydro's emergency preparedness capability and liaison with Provincial and Federal agencies. The Section identifies necessary activities to be performed in the area of emergency preparedness, identifies the required resources for these activities, coordinates their implementation, and reports the results. The Section coordinates the plans and procedures of all the response groups within the Corporation to ensure completeness, compliance with requirements, and consistency. The Section meets regularly with provincial and municipal authorities as a method to ensure coordination of the various organizational responses.

There are many departments within Ontario Hydro which have emergency preparedness responsibilities. The manager of each nuclear site and station is responsible for ensuring a response capability to protect workers, to mitigate any consequences of an accident, and to protect the public. To satisfy this responsibility, each facility has a response organization, detailed implementing procedures, the facilities and equipment to support response, and a training program for all response personnel.

A fundamental Ontario Hydro emergency response principle is that the on-site operating organization must be capable of implementing immediate response using on-shift staff. Such activities include personnel accountability, search and rescue, first aid and transport of injured personnel, fire fighting, regaining control of a damaged system, mitigating the consequences of an accident, emergency assessment and notification, on-site and off-site radiological surveys, protection action recommendations and the implementation of these protective actions. Each site and station trains shift personnel in these activities and holds routine drills and exercises to review the adequacy of the response capability and identify areas that can be improved.

Many other departments are involved in the development and maintenance of the emergency response capability. Technical support work is performed by Health and Safety Division, Design and Development Division - Generation, Research Division, Power System Operations Division, Technical and Training Services Division and Public Relations Division. Such activities include the development and maintenance of a dose assessment methodology for each station, an effective telecommunications system, training programs for Ontario Hydro, municipal and Provincial emergency personnel, a public education program, and an emergency public information program.

13.2.3 Plans and Procedures

Each Ontario Hydro emergency response organization has developed implementation procedures which allow exercise of its responsibilities as defined by the Corporation and within the Province of Ontario Nuclear Emergency Plan - Part I.

The nuclear sites and stations have procedures to provide worker safety, regain control of the affected system, and off-site response. These procedures, identify all immediate activities required to be performed by shift personnel.

Although the station procedures and shift organization are capable of performing all immediate activities, each facility has a management group which, upon notification, assembles and carries out specified functions. This group supports the on-site shift response by providing communication with external response groups, by acquiring required resources, by solving technical problems, and by reviewing actions and decisions. This management group also ensures that required support to the municipal organization is provided. The response group is normally in contact with the Corporate response organizations as well as the Municipal Control Group.

Ontario Hydro's Corporate emergency response organization provides a coordinated Corporate response to ensure Ontario Hydro's commitments to the province are met. In addition, this group supports and reviews activities and decisions at the nuclear site and station, and keeps the Executive Office apprised of on-site and off-site activities. This organization, called the Ontario Hydro Emergency Operations Centre (OHEOC), is headed by the Director, Nuclear Generation Division as the Recovery Manager, and includes experts in reactor safety, radiation protection and technical support. The areas for which the OHEOC provides support are in estimating future emissions of radioactivity, determining off-site consequences, providing technical support and resources, and providing information to the public through the news media. Procedures are in place for the OHEOC to perform its responsibilities.

One group which would be activated at the request of the OHEOC is the Design Core Group. This group consists of experts in the design and safety analysis of the Ontario Hydro nuclear stations, and is headed by the Director, Design and Development Division - Generation. This group would respond to specific technical problems identified by the station or the OHEOC, as well as identify the possible consequences of various response strategies and options. Procedures are in place for the Design Core Group.

Since an emergency at a nuclear station has Corporate wide ramifications, the Ontario Hydro Executive Office Team would be notified. This team would assemble at its operation centre, and provide Corporate direction during all phases of the emergency. The Executive Office Team would respond to both the immediate problem and the long-term implications. Areas of concern to this group would include not only the safety of Ontario Hydro employees and the public, but also the public perception of Ontario Hydro's response, the communications with senior Provincial and Federal politicians, communications to all Ontario Hydro employees on the status of the emergency and the capability of Ontario Hydro to supply electricity reliably. The group consists of the Chairman, President, Executive Vice-President - Operations, and appropriate Vice-Presidents. An operating manual provides the guidance for the group to carry out its responsibilities.

Ontario Hydro must interface closely with the provincial response organization (which has overall responsibility for off-site directives), the municipal response organization (which has overall responsibility to implement all protective actions and to make initial off-site decisions), the federal authorities, and the general public. Figure 13-1 summarizes the interfaces among the Ontario Hydro response organizations as well as between Ontario Hydro and external response organizations. The solid lines represent the flow of directives, while the dotted lines give the flow of information or questions.

FIGURE 13-1

EMERGENCY ORGANIZATIONS INTERFACE

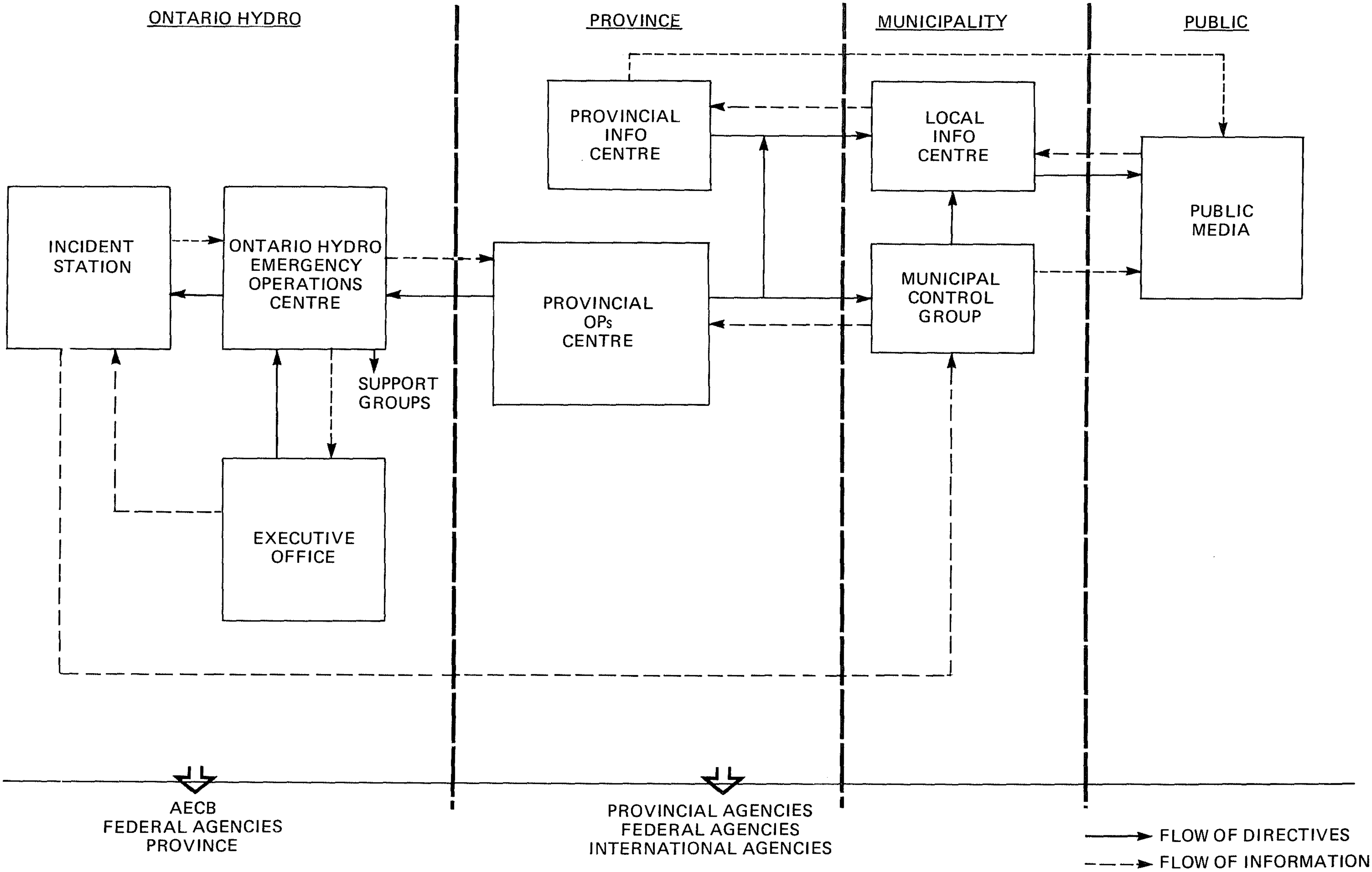


Figure 13-2 summarizes the notification process to inform all the response organizations.

13.2.4 Emergency Response Equipment

Because of the need for quick response capability as well as the need for reliable and accurate flow of information, Ontario Hydro has identified facilities for the response groups as well as dedicated support equipment.

Each nuclear station and site has the equipment necessary to perform the various on-site and off-site activities. Such equipment includes protective clothing, self-contained breathing systems, survey instruments, air samplers, two concentric rings of thermoluminescent dosimeters (TLDs), first aid equipment, firefighting equipment, in-station communication systems, off-site communication system, and an emergency survey vehicle. Each response group on the site has an identified response centre which contains all appropriate procedures and manuals, status boards, maps, and communication links.

The OHEOC has an identified facility which is equipped with maps for the Primary Zone of all facilities, plant and off-site status boards, appropriate procedures and manuals, a computer to perform dose assessments, and the necessary communication links.

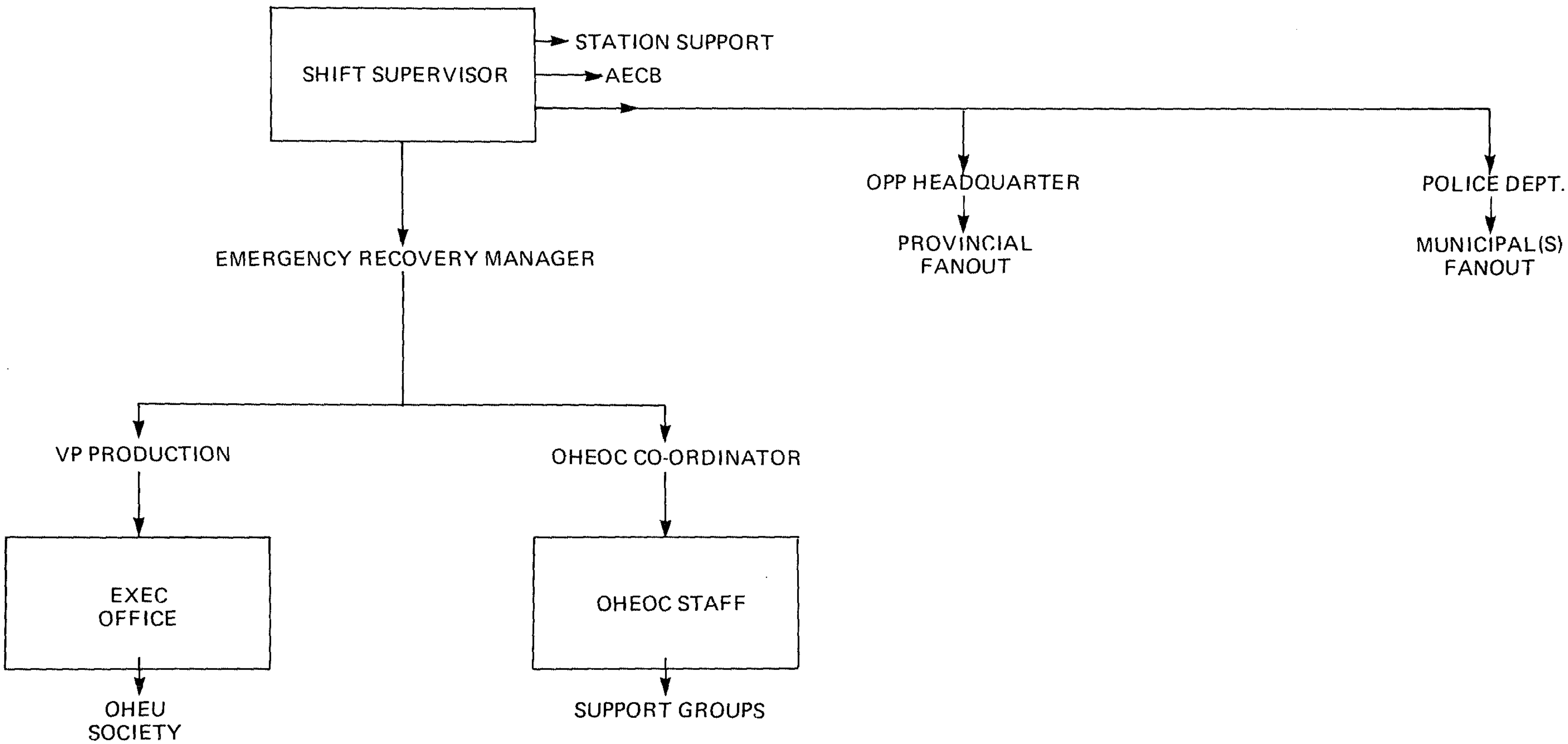
The Executive Office Team has an identified facility which is equipped with maps, plant and off-site status boards, appropriate procedures and manuals, television to monitor activities and press conferences at the Provincial Information Centre, and the required communication links.

Because of the various locations of Provincial, Municipal and Ontario Hydro response groups, there is a need for a reliable and secure telecommunication system. The present system consists of dedicated telephone lines (independent of the normal Bell system) between the OHEOC and the Executive Office Centre, the Design Core Group, the nuclear sites, the Provincial Operations Centre, the Provincial Information Centre, and the various municipalities. A central switchboard permits any centre to be in contact with another, as well as a conference system. Data, reports, and press releases are transmitted through a telecopier system to the various centres. The Ontario Hydro Centrex III system and the normal Bell system are used as a back-up to the dedicated system.

13.2.5 Dose Assessment

Dose assessment is the methodology used to predict the radiological consequences to the public. It may take many forms, use a variety of models, and combine different methods of calculation. Dose assessments are the basis for protective action decisions.

FIGURE 13-2
EMERGENCY NOTIFICATION



Ontario Hydro uses both a manual calculation method and computer programs to make dose assessments. The factors affecting dose assessments include the estimation of the source term, estimated time of release, release duration, release rate, atmospheric dispersion, and conversion of activity concentrations to dose.

The radiological source term is the type and amount of radioactive material as a function of time that will be released from the station. There are several methods to estimate the source term. The first method available is to use default values based upon the type of accident and status of the safety systems. This value can be refined by taking radiation surveys at predetermined locations, sampling airborne activity in certain locations, sampling process systems, and monitoring the radioactivity released through filters from containment.

The source term would likely include a small, prompt, short-term release due to containment overpressurization, followed by a holdup period, and finally a long-term release due to containment venting using the Emergency Filtered Air Discharge System (EFADS). The actual time of release depends upon the rate at which containment repressurizes. The dose assessment model uses available station parameters to estimate this repressurization time (see Figure 3-7 and Section 4.2.3.4).

The release duration refers to the time interval during which radionuclides are released from the station. Release rate is the rate at which radioactivity is released to the environment. These factors determine the total amount of radioactivity released to the environment.

The atmospheric dispersion of the radioactivity depends upon several factors such as the release height, wind direction, wind speed, and atmospheric stability. A computerized atmospheric dispersion model was developed by a Provincial working group for use in the Provincial Nuclear Emergency Plan. This model, Nuclear Emergency Atmospheric Dispersion Model (NEADM) consist of a puff release model and a continuous release or plume model. The puff model, AQPUF, is a Gaussian puff model which estimates the instantaneous concentration due to a puff release. The plume model, Continuous Release Dispersion Model (CRDM), uses an approach based upon growth of the plume.

To convert concentration to dose, approved dose conversion factors are used. These are based upon the principles of the International Committee on Radiological Protection (ICRP) in the documents ICRP-26 (Reference 3) and ICRP-30 (Reference 4).

Each station has a manual method to perform these dose assessments. As well, station-specific computer programs are being developed to provide increased accuracy, reliability, and speed. These programs are being developed in cooperation with the Province so that they can be used by the station, Corporate response groups and the Technical Group within the Provincial Operations Centre.

Dose assessment models are predictive estimates of the consequences of a release. Once a release has occurred, off-site monitoring is performed to provide actual measurement of the consequences of releases. This monitoring consists of two rings of thermoluminescent dosimeters located along the inner and outer boundaries of the Contiguous Zone to measure the integrated dose and formal survey teams to measure dose rate, concentration of both airborne and deposited radioactivity.

13.2.6 Protective Action Implementation

Once the consequences of the event have been estimated by means of dose assessment models and off-site radiological monitoring, a comparison is made with the provincial Protective Action Levels to determine the need for protective actions.

The Protective Action Levels (PALs) are summarized in Table 13-1. If the lower PAL is expected to be reached, the protective action is required to be implemented unless there are valid reasons for postponement.

TABLE 13-1

Protective Action Levels

	Lower Level		Upper Level	
	Effective	Thyroid	Effective	Thyroid
Sheltering	0.1 rem	0.3 rem	1 rem	3 rem
Evacuation	1 rem	3 rem	10 rem	30 rem
Thyroid Blocking	--	3 rem	-	30 rem
Food/Water Control	0.05 rem	0.15 rem	0.5 rem	1.5 rem

In the determination of the need for protective actions, previously received doses are added to conservative future dose projections. In addition, dose reductions from previously implemented actions are not taken into account.

Both the values of provincial PALs and dose projection methodology are more conservative than is common practice. For comparison, the International Committee on Radiological Protection recommends in ICRP-40 (Reference 5) a Protection Action Level of 5 rem to the whole body (lower PAL) and 50 rem to the whole body (upper PAL). For evacuation ICRP further recommends that a protective action is not justified below the lower PAL. Between the lower and upper PALs, action should be taken on an individual basis and should normally be implemented above the upper PAL. ICRP recommends the use of realistic estimates of projected dose and corrections for previously implemented protective actions.

As another example of Protective Action Levels, the National Committee on Radiological Protection recommends that potassium iodide (KI) pills not be distributed below a thyroid dose-saving of 25 rem (Reference 6).

Once a decision is made to invoke protective action, the municipality has responsibility to implement it. Ontario Hydro assists the municipality by providing radiological protection guidance at an Exposure Control Centre and contamination monitoring at Reception Centres. If thyroid blocking is a required action, Ontario Hydro would supply potassium iodide pills at the Reception Centres. Detailed procedures have been developed by the Ontario Hydro and the local municipalities. Responsible personnel are trained and tested periodically during exercises.

13.2.7 Public Education

An important element of successful emergency actions is the response of the public to protective action directions. The province, designated municipality, and Ontario Hydro have started a joint public education program around each Ontario Hydro nuclear site. The program consists of an information booklet distributed to each home and business within the Primary Zone, as well as an opportunity for individuals to have specific questions answered during a series of open houses.

The booklet includes definitions of nuclear emergencies, the notification process, the identification of response sectors, the procedure to implement various protective actions (including traffic control, sheltering, thyroid protection, and evacuation), special circumstances, and background information on radiation and reactor safety.

The public education program is an on-going process. It not only includes the annual issuance of the booklet with information seminars, but also involves shows on local cable television, articles in local papers, talks to interested organizations and opportunities for further information through any of the responsible organizations.

13.2.8 Emergency Public Information Program

The Province of Ontario Nuclear Emergency Plan - Part I identifies the objectives of the emergency public information program to be communicated to the affected public, provision of accurate information on the emergency to the general public and news media, and the assessment of the success of the information function. To carry out these objectives, the province has established a Provincial Information Centre and a Local Information Centre.

The primary role of the Local Information Centre is to communicate with the affected public regarding protective action decisions and directions. However, this centre also provides all emergency information until the Provincial Information Centre is established. Once the Provincial Information Centre is established, the Local Information Centre relays the general emergency information. The centre has spokespersons from the Province, Municipal Control Group and the nuclear site, along with the necessary support staff. This support staff includes writers, personnel to answer telephone inquiries, clerical, administrative, and security support.

The Provincial Information Centre, which is located at the Ontario Hydro head office complex, is responsible for the generation and dissemination of all general emergency information. Spokespersons from the Provincial Operations Centre and the Ontario Hydro Operations Centre (OHEOC) would communicate to the news media through routine press conferences. This Centre would also supply a writing service, media monitoring, as well as clerical and administrative support. Because this centre is the point for the dissemination of information to the general public, it is necessary for it to have contacts at the Provincial Operations Centre and the OHEOC. The Provincial Operations Centre includes an Information Group, which is responsible for providing the operations group with public perception feedback regarding the off-site consequences and protective action decisions. As well, this group relays appropriate information from the Provincial Operations Centre to the Provincial Information Centre.

The OHEOC includes a Technical Information Manager. This person is the link between the OHEOC (and thus the nuclear station) and the Ontario Hydro spokesperson at the Provincial Information Centre. As this link, the manager is the source of on-site information, such as reactor status and worker safety, and is the contact for assistance to the Ontario Hydro spokesperson.

The flow of information to the public is a critical component of the overall response capability. It not only has operational impact on the affected public, but also shapes the public's confidence in the management of the emergency. As a result, the emergency public information program is routinely reviewed by the NEPC, and needed improvements are identified and implemented in conjunction with the province.

13.2.9 Emergency Worker Training

Trained Ontario Hydro, municipal and provincial emergency personnel would be required to perform important activities in the event of an emergency at a nuclear station. Each nuclear station, therefore provides emergency response personnel with the necessary training and drills. This is done for both on-site and off-site emergency response activities. As well, personnel who participate in Corporate operations centres receive annual training and participate in exercises.

Ontario Hydro participates in the design and delivery of training programs to municipal and provincial emergency response organizations. Such training includes an overview of reactor safety, radiation protection and the emergency plans, as well as a detailed discussion on specific procedures. Such training normally has been part of the preparations for a provincial exercise. Because of the need for both a more formal and routine training program, Ontario Hydro is working with the province and emergency response groups in the development of training packages. These packages will make emergency response training easily accessible to those personnel who would respond in the event of a nuclear emergency.

13.2.10 Interfaces with External Agencies

Since a nuclear emergency would require provincial, municipal, and Ontario Hydro responses, close coordination and cooperation is required among all groups. This is accomplished through frequent meetings among these groups.

Ontario Hydro has participated in the development of municipal procedures, especially those procedures for which the Corporation provides significant support such as the Exposure Control Centre and Decontamination Centres.

Ontario Hydro reviews all provincial and municipal plans to ensure that its own plans and procedures are consistent. As well, the Corporation provides assistance to the Province in the development of their implementing procedures. For example, in the development of a computerized dose assessment program, provincial needs and procedures have been incorporated.

Ontario Hydro fully participates in the various provincial and municipal committees and working groups. In this way, the Corporation remains aware of future changes and requirements, as well as makes a contribution to the success of the groups.

13.2.11 Response to Transborder Accidents

Although Ontario Hydro has direct responsibilities in the event of an accident at one of its facilities, Ontario Hydro capabilities and resources are available to the province in the event of a nuclear accident outside the boundaries of Ontario.

The provincial response is summarized in the Province of Ontario Nuclear Emergency Plan - Part 8. Ontario Hydro contributes to this response at a level dependent upon the extent of the hazard. For an accident at a distant location, such as the Chernobyl accident, Ontario Hydro provided the province with any information it had of the accident, as well as the results of its environmental monitoring program.

Ontario Hydro has agreed to provide particular assistance to the province in the event of an accident at Fermi 2, which includes Essex County in its planning zone. This assistance includes the development of maps, procedures, personnel, training and equipment for the plume exposure control monitoring; procedures and trained personnel for the Exposure

Control/Decontamination Centres; trained personnel for the Local Information Centre; radiological protection training to municipal personnel, as well as any further need at the time of the accident.

Although Ontario Hydro would participate in a response to transborder incidents, it does so at the request of the Province of Ontario. The province, through MOSG, has complete responsibility for all off-site actions.

13.2.12 Exercises and Evaluations

Ontario Hydro annually reviews its emergency response capability. This is normally done through a combination of exercises and evaluations.

Each nuclear station has a number of emergency drills through the year, and at least one formal exercise. The Corporate response centres participate in two of these formal exercises. Municipalities may participate in these exercises. However, if they choose not to participate, their activities are simulated.

Ontario Hydro fully participates in provincial nuclear emergency exercises. This participation includes on-site and local off-site activities, Corporate response centres, and Ontario Hydro personnel who are members of the provincial response organization.

The results of each exercise are documented in an evaluation report and reviewed for relevancy at other stations by the Emergency Preparedness Section. Recommendations are documented and tracked on the Corporate Emergency Preparedness Tracking System.

In addition, there are periodic evaluations of the program or certain components of the program. These may be initiated by an event such as the Chernobyl accident, or as a follow-up to exercises, or by an external authority. In each case, the results of the evaluation are entered in the tracking system and monitored by the NEPC.

Through exercises and evaluations, the emergency response capability can continually be improved. Through the Corporate Emergency Preparedness Tracking System, identified improvements are being implemented and progress to completion monitored.

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Nuclear Safety: Public Attitudes, Consultation and Communications

14.0 NUCLEAR SAFETY: PUBLIC ATTITUDES, CONSULTATION AND COMMUNICATIONS

14.1 Introduction

Ontario Hydro has a commitment to communicate with the public and respond to public concerns in its policies, planning and operations. Objectives of the corporation in its relationship with the Ontario community include:

- Gathering information from the Ontario community on matters affecting Hydro.

- Considering the concerns and needs of the Ontario community in Hydro's decision-making.

- Developing a broad understanding among the Ontario community of Hydro's plans, policies and activities.

These objectives apply to Hydro's activities related to nuclear safety as well as many other aspects of Hydro's business.

Employees across the corporation, in their contacts with government agencies, customers, news media and the public, help communicate Hydro's commitment to public safety and increase the understanding of nuclear safety among the Ontario community. Some of these activities are required by legislation but Hydro's efforts go beyond these minimum requirements.

The Corporate Relations Branch has a particular responsibility for communicating with, and gathering information from the Ontario public. The branch provides a number of programs as well as providing communications and co-ordination services to Nuclear Generation, Design and Development and other divisions.

Activities to carry out Hydro's commitment to communication and consultation on nuclear safety matters are described in the following sections.

14.2 Public Attitude Surveys and Results

Ontario Hydro uses surveys and a variety of other sources to ascertain the concerns, expectations and evaluations of the Ontario public about Hydro's plans and activities. This information assists in understanding public feelings about nuclear power, public concerns and the topics on which the public wants information. The results of these studies are used to assist decision-making and to ensure that communications programs are topical and timely.

Ontario residents show a willingness to live with the current use of nuclear energy for electricity production. However, events at operating stations cause people to reassess the technology and to seek credible and reassuring information from 'third party' experts and the operator.

Public Views on the Use of Nuclear Power

Since Hydro began tracking public opinion on nuclear power in the mid-1970s, support for the current use of nuclear has remained essentially stable at 50-57 percent. Support for future nuclear development, however, is significantly lower at 30-40 percent (Figure 14.1). Events such as Three Mile Island and Chernobyl have not resulted in significant shifts in the public's confidence in nuclear technology, but rather have caused the public to seek reassurance on its safety. The views of those opposed to nuclear power tend to be reinforced by events such as Chernobyl.

Research has shown that support for the use of nuclear power in Ontario is based on three key factors: confidence in Ontario Hydro as an operator; familiarity with the technology; and availability of information. Support for nuclear power is also related to confidence in high technology generally. Those who support or oppose high technology usually have similar feelings about nuclear power.

Public Views of Nuclear Safety

Concerns about nuclear plant safety have been major factors in people's views on the future use of nuclear power in Ontario (Figure 14.2). The Ontario public generally seeks a higher level of safety from the nuclear industry than from other industries. Nuclear waste management remains a public concern but events such as Three Mile Island and Chernobyl have resulted in higher levels of concern for nuclear plant safety. This has resulted in a heightened provincial and national view that public concern about nuclear safety is justified.

There are variations among the population in views about the risks of nuclear technology. Opponents of nuclear power tend to be more concerned with the probability of an accident. The balance of the population is more concerned with the potential consequences. Eighty percent of Ontario residents are more concerned about human error, causing an accident than about a failure in the technology. Only 23 percent believe that the consequences of an accident at a nuclear plant could be contained while 68 percent believe that the consequences would be felt across the province.

In evaluating the safety of nuclear power plants, Ontario residents concentrate on three key factors in order of priority:

- Ontario Hydro's operating safety record;
- Expert and 'third party' assessments of the design safety of Ontario's nuclear plants;
- Overall technical expertise of Ontario Hydro.

Ontario Hydro as a Nuclear Operator

Ontario Hydro has a public image of taking its public health and safety responsibilities seriously. About 70 percent of the Ontario community have confidence in Hydro's ability to contain any risk or serious accident at a nuclear plant and thus minimize the danger to those outside the plant. This view is endorsed more strongly by people living near nuclear plants. The majority of residents in these areas feel safe and believe that Hydro and other organizations are concerned about public safety and demonstrate this concern through operating and emergency procedures. For example, Pickering residents speak positively of Hydro's handling of the tube failure and its ongoing approach to community consultation.

Information as a Factor

A higher education level and familiarity with a nuclear plant continue to be key factors in nuclear support. While few residents have sought information on nuclear power directly from Hydro, 83 percent have found the information interesting and informative. The response of those who have visited a plant is similar.

The seriousness of nuclear technology as an issue causes people to use a variety of information sources. The scientific community and Ontario Hydro are seen by Ontario residents as the most credible sources of information on nuclear power plant safety.

Determining Safety

Determining safety and acceptable risk are seen by the public as a shared responsibility of government, the scientific community and Ontario Hydro. However, public perception of government's capability to ensure nuclear plants are built and operated safely has eroded to its lowest level in eight years (from 56 percent nationally to the current 47 percent).

Women and Nuclear Safety

Public attitude studies show that women tend to have more entrenched and negative feelings towards nuclear power. For example, opposition to the current and future use of nuclear power is significantly higher among women than men. Women also tend to have less confidence in high technology and thus express lower confidence in the design and operating safety of Hydro's nuclear plants. Women are generally more concerned than men about the extent and seriousness of the potential effects of an accident at an operating nuclear plant. They are also more firm than men in the view that the risks of nuclear outweigh any benefits of lower rates. Events such as Chernobyl have had more effect on women's attitudes to the safety of Ontario Hydro's nuclear plants (54 percent of women have changed their view as compared to 36 percent for men). It also resulted in women having a higher level of concern that a similar nuclear accident could happen in Ontario.

While women tend to be more dissatisfied with the information on nuclear power provided by Ontario Hydro, studies show that they have made less effort to seek out information on nuclear power or to visit a nuclear station.

14.3 Public Consultation

Since the early 1970s, public involvement has been an integral part of Hydro's planning process, mainly for new generation and transmission facilities. More recently, public consultation has been extended to other areas such as long range planning and tritium recovery. Several Hydro Forums were held in 1985 to discuss the general concerns of customers. Consultation programs provide opportunities for discussion between Hydro and the Ontario community on a wide range of issues.

In 1984, Ontario Hydro began the Demand/Supply Option Study to examine ways of meeting the province's electricity needs in the 1990s and beyond. Options to increase the supply and to reduce the demand for electricity are being examined. Public consultation has been an important part of the development of a strategy to guide planning for the next 10 to 20 years.

In 1986, consultation programs were undertaken with provincial organizations, community leaders and municipal utility representatives. Of 120 provincial organizations invited to participate, 58 took part in meetings and 36 submitted briefs. A regional program included 13 discussion meetings across the province with more than 300 people representing a variety of community interests. The municipal utility program included 10 meetings with more than 300 officials representing more than 100 utilities and five meetings with Hydro's direct industrial customers.

The consultation programs were developed to gain an understanding of public concerns, priorities and values regarding planning for future energy needs and the supply and demand options. Consensus was not sought but rather the range of opinion among interests and across the province. Although views about nuclear safety were not specifically requested, many participants expressed opinions about the nuclear option.

The values, attitudes and opinions of Hydro's customers were also sought through a survey undertaken in early 1986. Interviews were carried out with 1,200 residential, 200 commercial and 200 industrial customers.

Opinions about nuclear power among participants in the consultation programs and survey ranged from strong support to conditional support to absolute opposition. As a means of supplying future energy needs, nuclear generally fell behind hydraulic and alternative technologies but ahead of coal and oil in public preference.

The reasons expressed for support of nuclear energy included its low cost, cost-efficiency, less environmental impact, and its contribution to the Ontario economy. Among the reasons for opposition to nuclear power, safety issues were the key concern. Public safety concerns included the views that nuclear plants are unsafe or dangerous, the possibility of accidents, and the potential for environmental and health impacts of nuclear waste disposal. Other reasons for opposition included the high capital costs and the possible connection with military use of nuclear technology.

The results of the consultation programs have been incorporated in a draft planning strategy which is currently under internal review. There will be further public consultation and government review before the strategy is approved.

14.4 Public Communications on Nuclear Safety

The availability of information is a major factor in public understanding about nuclear safety. Ontario Hydro is committed to providing accurate, understandable and timely information to meet the needs of the Ontario public. A wide range of methods and materials are used to inform the public about all aspects of nuclear energy including nuclear safety. Some programs deal mainly with nuclear safety while others include safety-related components. Information is provided directly to the public, through the media and through information provided to governments, educators and others.

Programs dealing mainly with safety topics include:

- Communications with the public and the media regarding emergency preparation exercises and procedures, and communications in the event of an emergency. These programs are described in detail in Section 13.

- Reporting significant occurrences at nuclear facilities to the public through the news media. These include events such as the Pickering tube failure which do not involve an off-site emergency. Hydro's policy is to report to the media in a timely manner all occurrences which may be important to the public.

- Co-ordination of an extensive communication/training program on the transportation of radioactive materials. This involves training police and firefighting forces throughout the province as well as making presentations to municipal councils and the public.

Programs which have a significant nuclear safety component include:

- Production and distribution of materials such as pamphlets, films, video-tapes and displays on safety-related topics such as reactor and safety systems design, operator training, radiation and emergency planning.

- An education program involving teacher aids, films and a news magazine for teachers and students of elementary and secondary schools. The program covers a wide range of energy topics including nuclear safety.

- An outreach program for schools and the public in the neighbourhoods of nuclear generating stations. In 1986, more than 72,500 people participated in outreach programs. Community activities also include newsletters, open houses and participation in community shows, fairs and events.

- Information centres which are open year-round at the three major nuclear stations. Many of the displays at the centres are safety-related. In 1986, more than 69,600 people visited the centres. Site tours are also available.

- A speakers' bureau program in which technically-qualified staff are available for presentations or participation in debates. Nuclear topics, principally safety, account for about 50 per cent of the speaking engagements (250 of 500 engagements in 1986).

- A public enquiry service which responds to letters and telephone enquiries from the public on any subject including nuclear safety. Of the more than 3700 enquiries in 1986, almost 500 concerned nuclear matters.

- A Public Reference Centre where documents relating to Hydro's nuclear program are available. These include technical documents such as safety and commissioning reports and documentation of hearings and official enquiries.

Ontario Hydro is also active in the communications programs of other organizations which provide information on nuclear safety. For example, Hydro produces and distributes an Energy and Ethics Newsletter for the Canadian Nuclear Association. The newsletter has a readership of more than 2500 in 15 countries.

The Corporate Relations Branch maintains a media relations unit for informing the media on any aspect of nuclear safety. Corporate Relations staff are also located at the nuclear stations to provide local services. In addition to providing occurrence reports and communications about emergency matters, Ontario Hydro provides:

- Media open houses at nuclear stations. Station tours are provided to media representatives.

- Background information, stock film footage and TV programs.

14.5 Communications with the Government

An important component of Hydro's commitment to nuclear safety is the development of a broad understanding by governments whose responsibilities may be affected by system operation. Hydro provides information on nuclear safety to all levels of government including: elected officials; ministries responsible for occupational health and safety, environmental protection and emergency planning; and municipal governments, fire and police forces in the area of Hydro's facilities.

Hydro often provides information to governments or their agencies to meet legislative requirements, or as a result of public hearing or formal agreement. Extensive informal arrangements have also evolved through convention and in response to governments' interest.

At the federal level, legislative requirements include extensive monitoring, inspection and reporting to the AECB. These requirements are discussed in detail in Section 1.

While the federal government has jurisdiction through the AECB over nuclear facilities, the Ontario Government has a keen interest in the operation of the facilities. The Ontario Government is kept informed on a regular basis through:

- periodic briefings on aspects of station design and operation;
- reports prepared for the AECB;
- reports prepared for the Ontario Government (e.g. Annual Report of the Nuclear Integrity Review Committee);
- site tours;
- informal procedures to ensure that various government ministers and senior officials are advised of significant events at nuclear facilities in a timely fashion.

The Ontario Government has the lead responsibility in developing nuclear emergency plans to ensure public safety in the event of a major incident at a nuclear facility. The Emergency Plan, dated June 12, 1986, was developed by the Ontario Solicitor General's office in consultation with Ontario Hydro, the AECB, AECL, 11 other ministries and the municipalities adjacent to Ontario's nuclear generating stations. The plan details responsibilities for emergency preparation, and the procedures to undertake in the event of an off-site emergency.

Programs are also in place for municipalities, local fire, police and emergency response groups in the vicinity of nuclear stations or along transportation routes for radioactive materials. These programs, discussed in Section 13, describe possible events at stations or road accidents involving nuclear materials and Hydro's emergency response procedures. On-going liaison with local officials in the vicinity of nuclear stations includes providing information on nuclear safety matters on a regular basis.

14.6 Public Enquiries and Hearings

Ontario Hydro has appeared before a number of government hearings, legislative committees and enquiries focussing on aspects of Hydro's nuclear operations. Public hearings provide an opportunity for government and the public to become better aware of Hydro's nuclear operations. These hearings also provide Hydro with an insight into government and public concerns regarding nuclear safety and ways in which safety can be improved.

Public hearings related to nuclear safety in which Hydro has participated include:

- Advisory Committee on Energy (1971-73) examined Ontario's energy policies. In 1973 the Committee stated that although the CANDU system is reputed to be one of the safest, there is a need for "close and continuous surveillance."

- Task Force Hydro (1971-73) examined many aspects of Ontario Hydro's operation. In 1973, the Task Force recommended that Hydro initiate a "major and sustained public information program related to nuclear power generation."

- Royal Commission on Electric Power Planning (1975-81) examined a wide range of Hydro's plans and activities including the use of nuclear energy. Regarding nuclear safety, the Commission concluded that "within reasonable limits, the reactor is safe."

- Select Committee on Hydro Affairs (1978-81) examined nuclear safety, load forecasting, uranium contracts and heavy water plants. The Committee found that the chance of a very serious accident occurring in any single reactor is extremely small and that "the reactors are, therefore, 'acceptably safe.'"

- Standing Committee on Resources Development (1979) examined problems with Pickering B boilers.

- Standing Committee on Public Accounts (1984) examined Darlington, the Pickering retubing and mothballing of old plants.

- Select Committee on Energy (1985-86) examined the need for Darlington and the Demand/Supply Options Study. Recommendation 3 of the Committee lead to the formation of this Commission.

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FIGURE 14-1

SUPPORT FOR PRESENT AND FUTURE USE OF NUCLEAR POWER

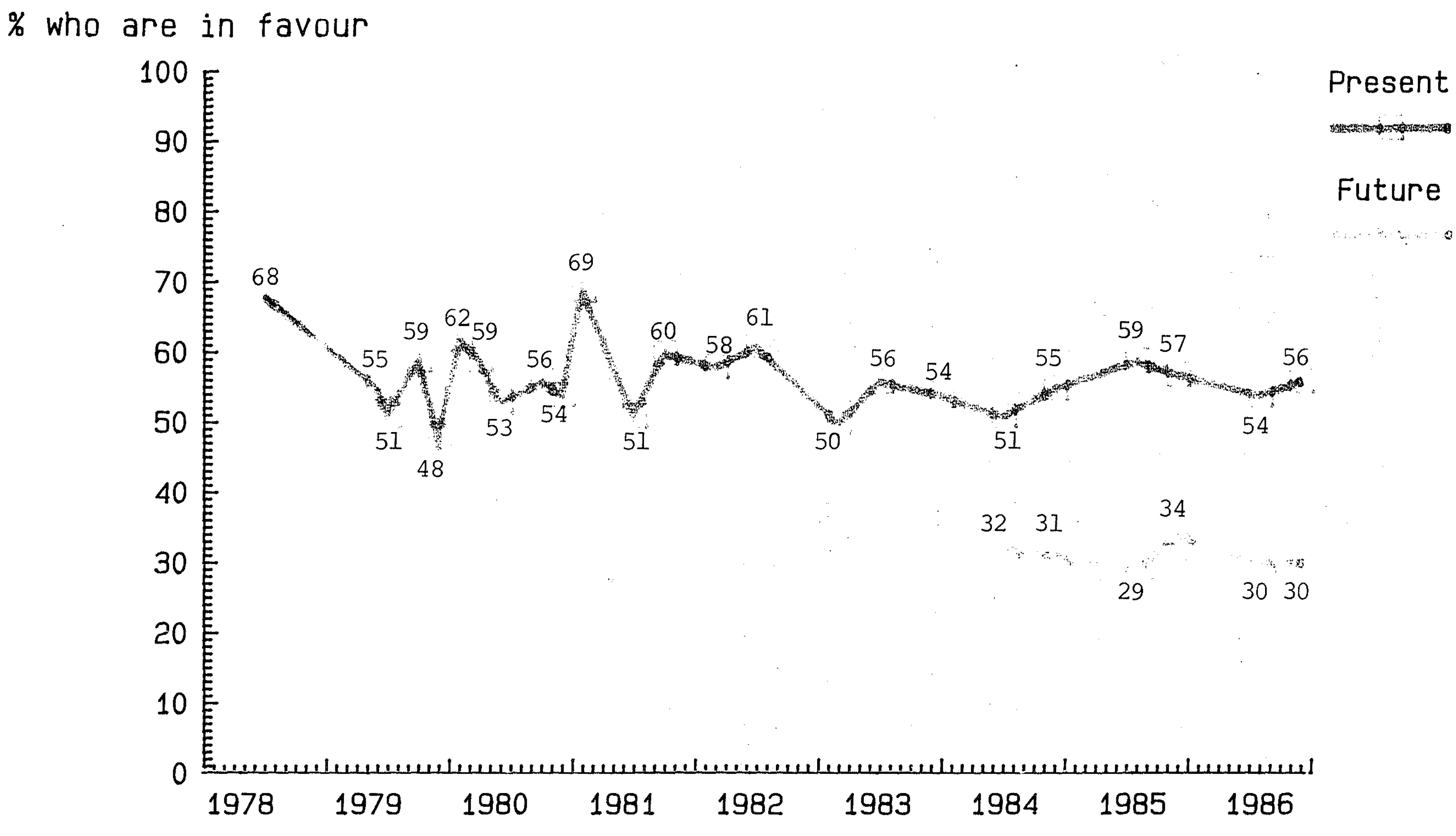
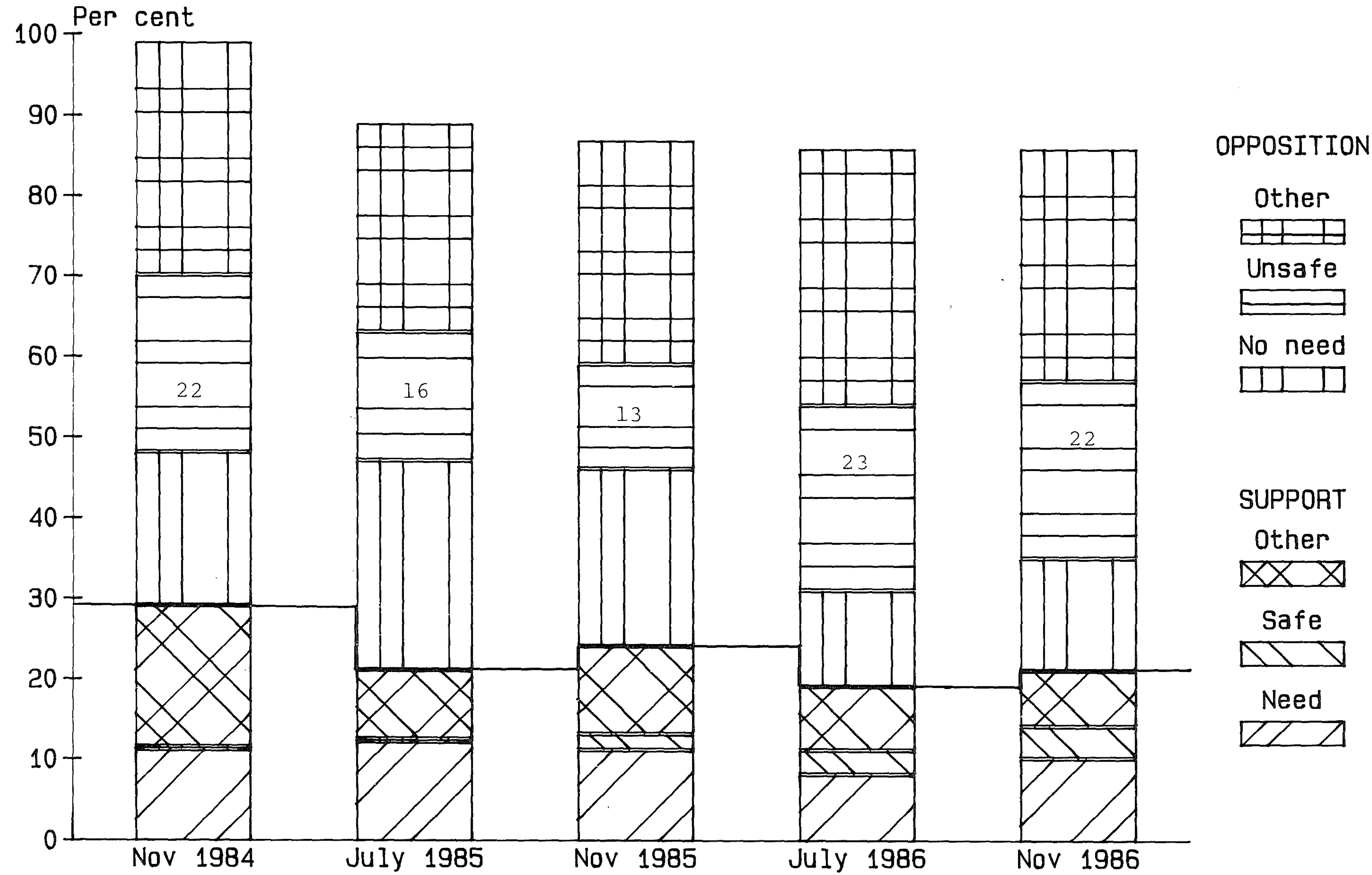


FIGURE 14-2
BASIS OF SUPPORT OR OPPOSITION TO FUTURE NUCLEAR POWER



November 1986

Inherently Safer Reactors

15.0 INHERENTLY SAFER REACTORS

This section will summarize the objectives of "inherently safer reactor" and will describe very briefly some of the concepts being developed. This should not in any way be interpreted as Ontario Hydro endorsement of the need for such reactors. The safety record of existing nuclear power reactors, and Ontario Hydro's CANDU reactors in particular, is indeed an enviable one when viewed objectively against other industrial endeavours.

There are several existing, small scale, experimental reactors which have inherently safe features but it should be stressed that these are nowhere near the commercial development stage and their economic viability for such a role is far from certain.

"Inherently safer reactors" are being designed to be able to shut themselves down and continuously cool their cores in the event of an accident, without having to rely on external power, mechanical devices or human operators for an extended period of time. Compared to the current reactors in operation, the inherently safe reactors theoretically provide a larger safety margin by allowing considerably longer 'grace period' following an accident before external interventions are required to limit the accident consequences.

Current nuclear power plants achieve their safety goals by incorporating an array of engineered safety systems, some designed to prevent accidental core overheating while others to mitigate the consequences of a damaged core. Some of these safety features, such as the negative pressure containment system in Ontario Hydro's CANDU stations, are virtually passive or self-actuated. But many others, such as emergency coolant injection and shutdown systems, must be activated, either automatically by electromechanical devices or manually by operator intervention. Multiple backup systems are used to reduce the probability of a major radioactive release to an acceptably low level. The design intent of the inherently safer reactors is that they not be dependent on electromechanical equipment nor operator actions to meet their safety objectives. Instead they utilize passive safety features such as natural convection or thermal radiation to prevent core overheating. Inherent core characteristics such as negative power and temperature coefficients of reactivity are utilized to provide self-regulation and self-stabilization in the event of an accident.

The following is a brief description of the inherently safe reactors being developed. Although they use different coolants and core cooling processes, they do share some common safety features such as:

- a high reactor vessel surface to fuel volume ratio to improve the heat transfer process.

- lower reactor output. The intent is that commercial sized units will be made up of modules each of which has a maximum power of up to 200 megawatts electrical.

- a larger reactor vessel with considerably larger coolant inventory for water cooled reactor design.

- larger negative temperature coefficient and power coefficient, i.e., as the core heats up, its properties change to decrease reactivity and limit the fission process.

PIUS

PIUS (Process Inherent, Ultimately Safe) is a water cooled reactor developed by Asea-Atom of Sweden. In a PIUS design, the entire reactor core, and coolant systems are immersed in a giant pool of pressurized water. The pool and the coolant circuit are hydraulically connected at both the upper and lower interfaces (see Figure 15-1). Because the hot coolant lies above the denser pool water, density locks are formed at the interfaces to prevent mixing of the coolant and pool water. The pool water contains a large amount of boron, a neutron absorber that will cut off the fission process if allowed to flow into the core via the interface.

Under normal operating condition, the circulating coolant, separated from the borated pool water by the density locks, transfers the heat from the core to the steam generator. If, for whatever reason, the coolant flow is disrupted, the pressure balance at the interfaces is disturbed and the borated pool water will flood the core by natural convection and shutdown the reactor. The large volume of water in the pool serves to absorb the decay heat of the core and guarantees approximately one week of 'grace period' before external intervention is required to ensure the continuation of shutdown cooling.

The density locks are formed when the buoyancy force on the coolant and the frictional force in the core balance each other. In other words, the pressure drop across the reactor core, is equal to the hydrostatic pressure of the cooler pool water. With the density locks formed at the interfaces, cross flow of the pool water and coolant are prevented. If a failure of the coolant pump occurs, the pressure balance is disturbed and the borated pool water will flow into the core via the interfaces to shutdown the reactor and cool the core.

There has not been any PIUS prototype built so far. However, in Vasteras, Sweden, a test apparatus built with all the essential parts of a PIUS system has been successful in confirming the inherently safe properties of the primary system.

Pebble bed - HTGR

HTGR (High-temperature gas-cooled reactor), designed by Interatom GmbH of West Germany, uses helium gas as coolant. Pressurized helium gas is pumped down through the core and passes on to a steam generator. Hundreds of thousands of fuel pebbles, small graphite spheres with fuel particles embedded inside, make up the reactor core. The pebble-bed reactor core and its surrounding graphite reflector are enclosed in a small reactor vessel (see Figure 15-2a). New fuel pebbles can be continuously fed into the core from the top while spent fuel pebbles are discharged from the bottom to allow on power fuelling. A HTGR would produce about 100 megawatts of electricity. Several modules of the HTGR can be combined to form a full-size power plant.

In the event the helium flow is cut off, the fuel pebbles inside the rector vessel will begin to heat up. With its power coefficient and temperature coefficient of reactivity both negative, the core will stabilize by itself due to negative feedback on the neutron flux. The temperature of the fuel pebbles will also stabilize by radiating heat to the surrounding reactor vessel. The small reactor vessel, designed to have a high surface-to-volume ratio, readily transfers the heat from its outside walls to the surrounding environment by natural air convection and thermal radiation, and acts as a heat sink for the reactor core.

The fuel pebble is a graphite sphere about the size of a tennis ball (See Figure 15-2b). It is composed of an outer fuel-free graphite shell with a wall thickness of around 5 mm. Inside the graphite shell the fuel is contained in the form of coated fuel particles embedded in a graphite matrix.

The fuel pebbles can withstand temperature of up to 1600°C without failure. The graphite inside the reactor vessel and inside the fuel pebbles can withstand temperature of well above 3000°C without losing its structural strength. With such high temperature resistance of the fuel pebbles and the core structural materials and the availability of the reactor vessel as a naturally cooled heat sink, the risk of a core meltdown in a HTGR is completely removed.

IFR

The IFR (Integrity Fast Reactor), developed by the Argonne National Laboratory in the United States, is a small breeder reactor design using liquid sodium as coolant. The excellent thermal properties of the liquid sodium allows the primary coolant system to operate at atmospheric pressure with a small coolant inventory.

The IFR, like the PIUS, has its reactor core, coolant pumps and heat exchanger submerged in a pool, but of sodium. A small portion of the sodium is pumped through the core to the heat exchanger (see Figure 15-3). If the coolant pumps fail, the pool will flood the core by natural convection, stopping the fission reaction and cooling the core.

PRISM

PRISM (Power Reactors Inherently Safe Module), designed by General Electric Company of the United States, is another version of the IFR design. In a PRISM reactor, the heated coolant is circulated to a heat exchanger located outside the reactor vessel (see Figure 15-4). If the coolant flow is interrupted, the sodium continues to cool the core by transferring heat to the reactor vessel. The small reactor vessel in turn radiates heat to the surrounding containment vessel, which is cooled by naturally circulating air around it.

SLOWPOKE

SLOWPOKE (Safe Low Power Critical (k) experiment) is a small low temperature water cooled reactor designed by Atomic Energy of Canada Limited with a thermal output capacity ranges from kilowatts to 2 megawatts. The SLOWPOKE reactor is more suitable for space heating applications rather then electricity production. The reactor consists of a small reactor core located at the bottom of a large pool of water at atmospheric pressure. The coolant is circulated by natural convection, thus eliminating the requirement of circulating pumps in the primary system (see Figure 15-5). The inherent safety features of the SLOWPOKE reactor include negative temperature and void coefficients of reactivity, so that heating or boiling of the coolant-moderator causes the reactivity to decrease. Such design provides inherent protection against reactivity transients, removes the requirements for backup safety devices and allows the reactor to be operated unattended.

OTHERS

Other advanced light water reactor designs under development also attempt to include some of the intrinsically safer features. Efforts are made to reduce the complexity of the designs and the dependence on electromechanical equipment and operator actions. Reactor core power density is lowered to improve thermal margins. Passive safety features such as natural heat transfer processes and negative core reactivity coefficients are adopted where possible. However, much development and testing are required before a commercial size intrinsically safer nuclear power plant can be put in operation.

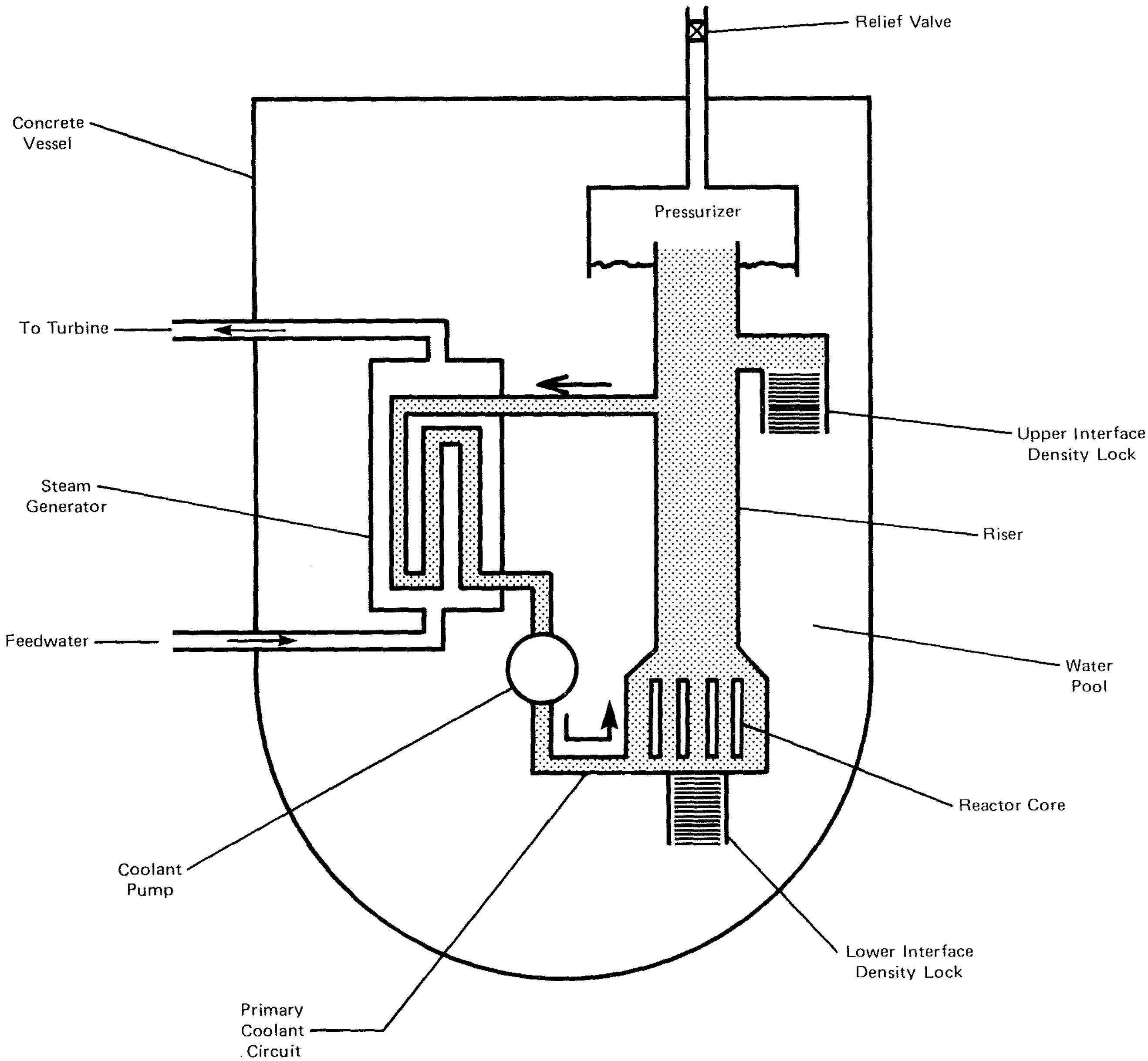


FIGURE 15-1
PIUS

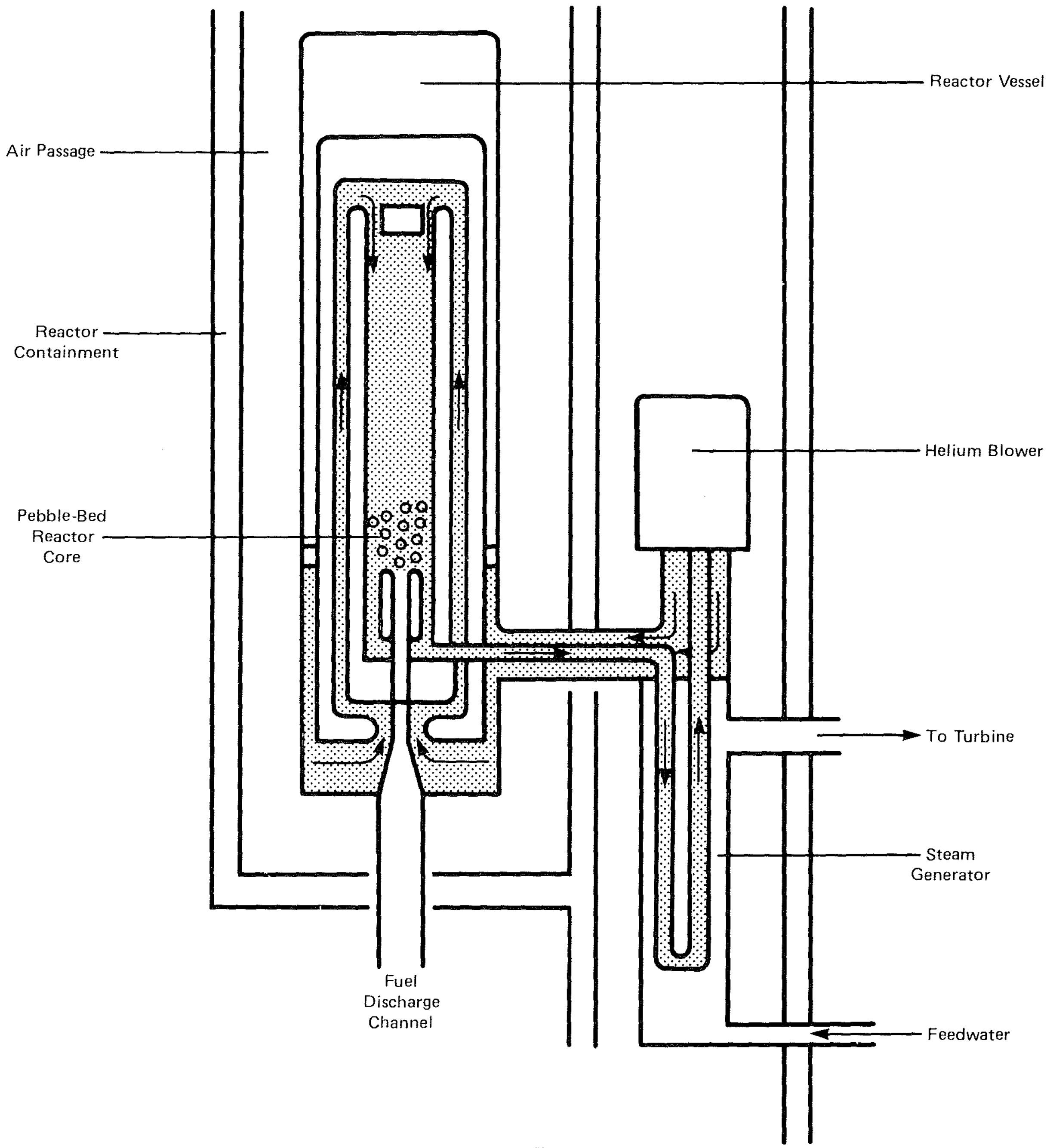
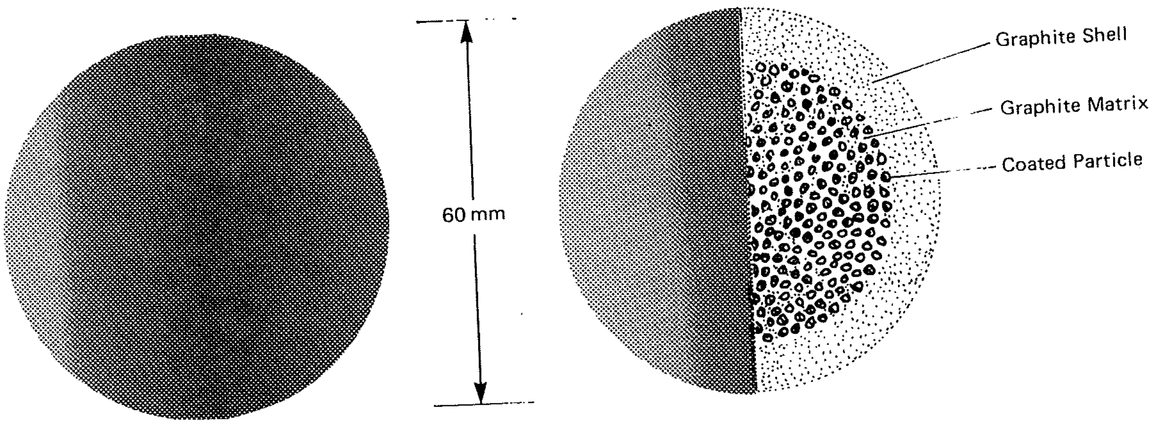


FIGURE 15-2a
Pebble-Bed HTGR



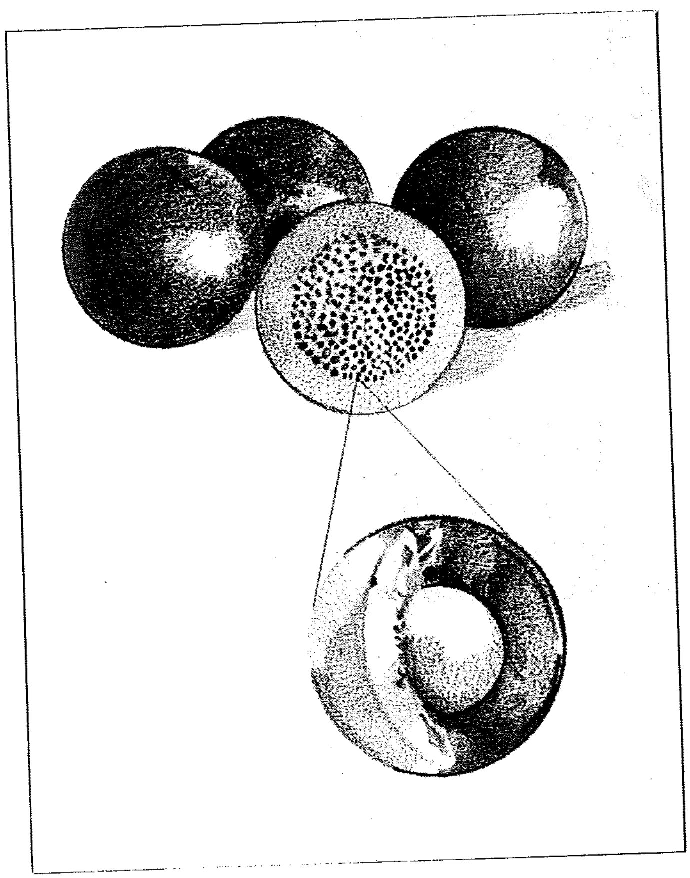


FIGURE 15-2b
Pebble-Bed HTGR Fuel Element

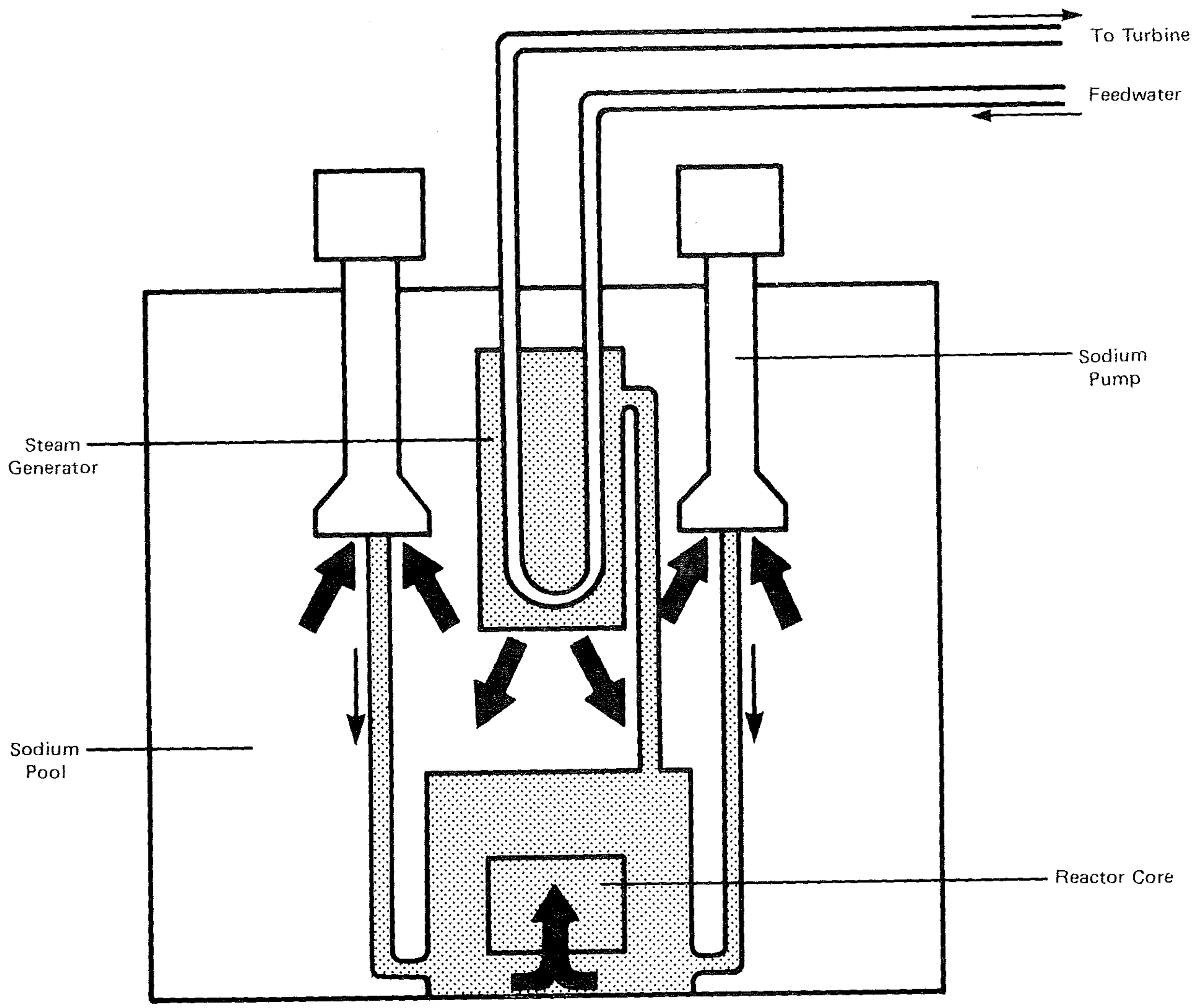


FIGURE 15-3
IFR

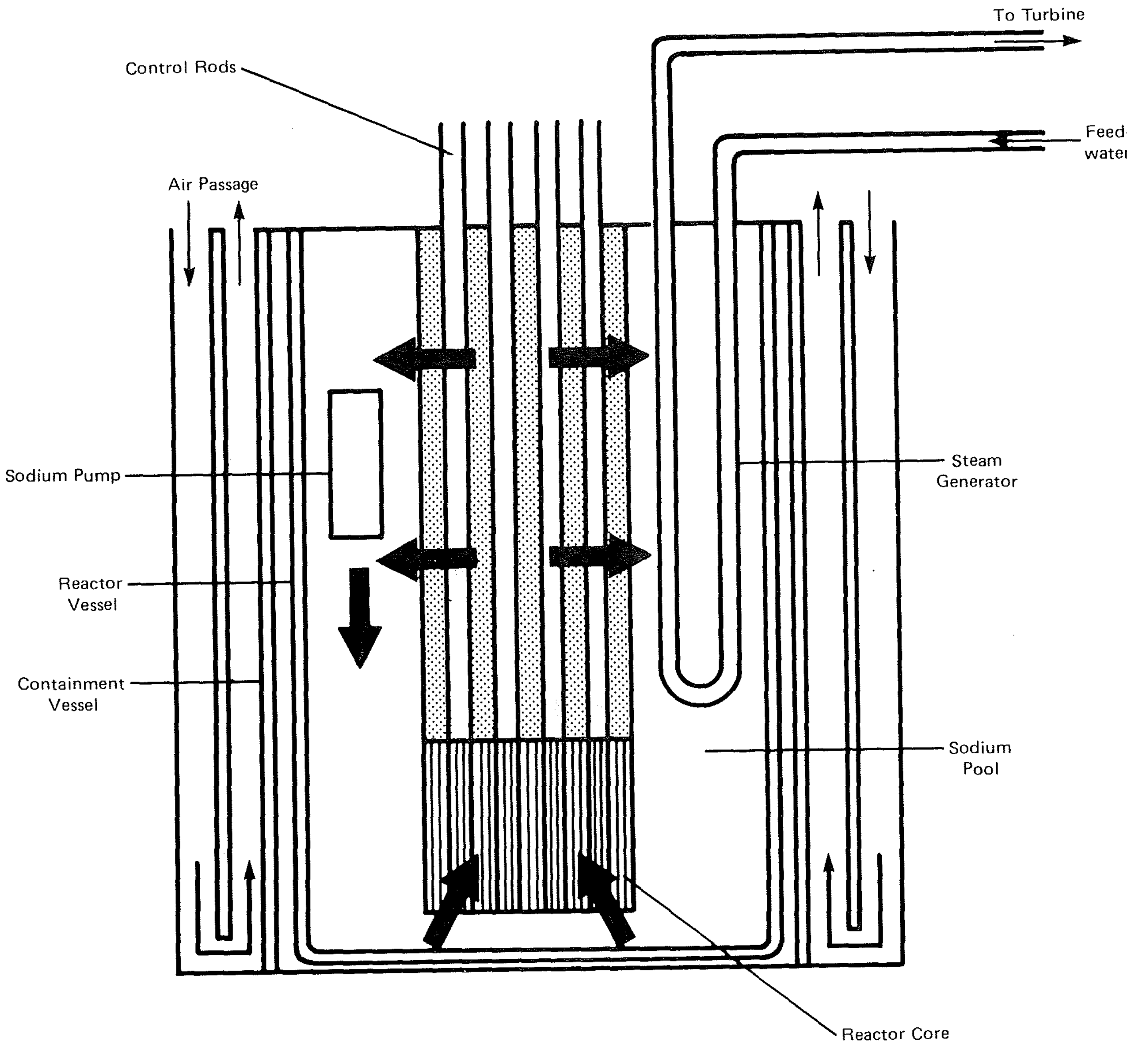


FIGURE 15-4
PRISM

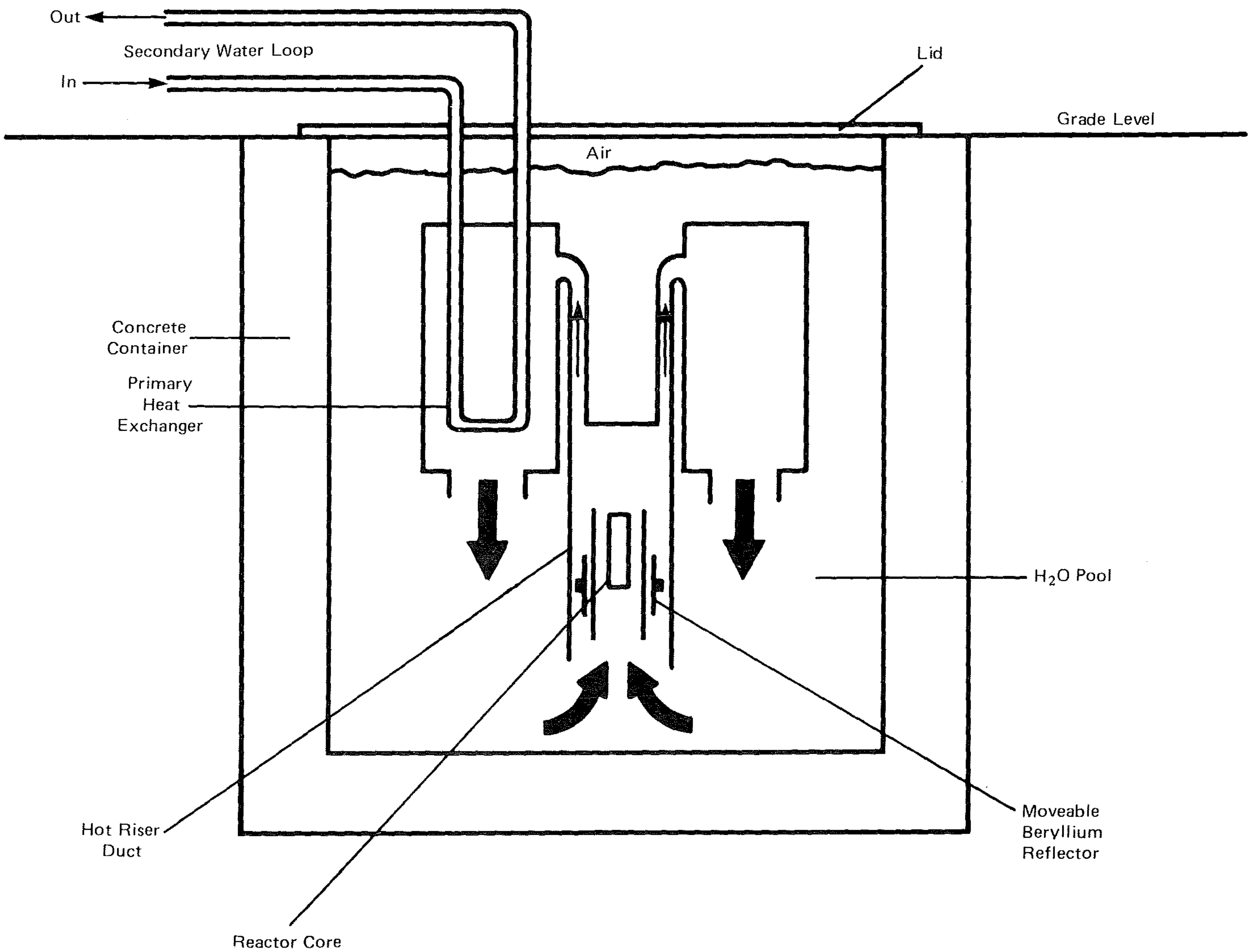


FIGURE 15-5
Slowpoke Heating Reactor

Nuclear Station
Ageing and Safety

16.0 NUCLEAR STATION AGEING AND SAFETY

16.1 Introduction

Production reliability, as well as public and worker safety, demands that unexpected failure be accommodated and the probability of such failure be minimized. Planning for nuclear station ageing is an integral part of Ontario Hydro's approach to ensuring high levels of safety and reliability.

Materials and components age and unless replaced or refurbished, will fail. Unexpected failure are therefore, pre-empted at the design stage by specifying materials and components in accordance with internationally accepted codes and standards. During operation, inspection and maintenance programs are established based on the known performance of the relevant materials or components under the relevant operational conditions.

16.2 Ageing Phenomena

With one exception, ageing mechanisms in nuclear stations are no different from those found in fossil fired power stations or chemical plants. Systems and components are affected by factors such as operating cycles (fatigue and wear out); temperature, pressure and chemistry variables; vibration; flow erosion and corrosion. These mechanisms are well documented in national and international codes and standards which formulate rules and recommendations for materials selection, allowable stress levels, lifetime corrosion allowances and other factors. In Canada, the ASME Boiler and Pressure Vessel Codes are used for nuclear components as well as CSA standards specific to CANDU plant and component design and quality assurance.

The ageing phenomena unique to nuclear generating stations are those related to the effects of neutron bombardment and gamma radiation on components and materials in, or in close proximity to, the reactor core. The reactor structure and its components and proximate systems (such as the fuelling machines) are subject to these phenomena in addition to conventional ageing mechanisms.

For example neutron bombardment leads to creep (growth/sag) of fuel channel components as well as an increase in strength at the expense of ductility. The uptake of hydrogen/deuterium from the heat transport coolant can lead to a decrease in ductility of fuel channels at "cold spots" or at points of high local stress where delayed hydride cracking can occur. (See Pressure Tube Performance, Section 9). Hence, reactor core components have always received special attention in research and development programs over the last 40 years in Canada (at WNRE/CRNL and Ontario Hydro), as well as throughout the world.

16.3 Manufacturing and Construction Activities

Ontario Hydro inspectors perform inspections at various stages of manufacturing and construction to ensure the highest level of quality assurance, consistent with the design intent. Various inspection techniques are used to ensure that as-built components meet the required level of quality and hence performance. For example, all nuclear Class 1 welds are radiographed to ensure that no defects are present. The materials of construction for the reactor components are specified on the basis of experience, and past performance, to ensure minimum maintenance or repair over the life of the station. Rigorous construction and installation standards ensure that a high quality product is handed over to Operations for commissioning.

16.4 Pre-operational Activities

Pre-operational activities provide an opportunity to develop, test and verify various surveillance, monitoring and periodic inspection programs which will later allow operators and support staff to monitor component performance and detect degradation, age-related or otherwise.

Critical components, such as the fuel channels, require dedicated pre-operational phase programs to benchmark the quality status of such components for periodic inspection. Such benchmarking includes various dimensional measurements, inspections of areas subject to corrosion or wear and inspection of areas subject to severe service conditions. This inaugural inspection provides an essential framework for establishing follow-up or periodic inspection programs during the life of the station. The activities normally carried out in the process of commissioning provide a substantial baseline database on the initial status of the components. Collection, analysis and recording of data for later use in the inspection of system integrity and assessment of ageing of critical components is a vital pre-operational activity in Ontario Hydro.

Post-operational design review provides an opportunity to carry out a design review after selected systems have been commissioned and operated for some time (about two years after the in-service date of the last unit). The various commissioning reports and in-service reports, together with Significant Event Reports (SERs) and other related documents serve as the input data package for such a review. Together with engineering phase design reviews, and the results of inspections during the manufacturing process, these provide an effective mechanism to discern early indications of component degradation.

16.5 Operation Activities

Inspection and maintenance programs are integral to the operation of Ontario Hydro's nuclear generating stations. These programs are designed to identify, and to correct for, the effects of ageing of system components.

For each nuclear generating unit, a Periodic Inspection Program (PIP) is established which conforms to the requirements of CSA Standard CAN3-N285.4-M83, Periodic Inspection of CANDU Nuclear Power Plant Components. Periodic Inspection is a jurisdictional requirement and is monitored by the AECB at all stages of its preparation and implementation throughout the life of the station. The purpose of the Periodic Inspection Program is to provide the necessary assurance that the likelihood of system and component failures has not increased since the plant was placed in-service.

In addition to the N285.4 requirements, the containment structures and containment components are inspected and tested. These inspection and testing activities are also reported to the AECB.

AECB licensing requirements demand that the special safety systems of the nuclear stations, e.g. (shut off rods, liquid poison injection, emergency coolant injection and containment) meet very high standards of reliability and availability. This applies not only to the components themselves but to the entire system of instrumentation, fault detection, electrical supply and other services required for the proper functioning of these systems. To achieve the required levels of safety performance, there is a program of continual testing of all components of the special safety systems with adjustment, maintenance or replacement, of any component that does not meet its individual performance target. This program ensures any degradation in special safety systems, whether due to ageing or other mechanisms, will be discovered promptly and rectified.

Ontario Hydro also performs In-Service Inspections (ISI) on critical nuclear components, such as pressure tubes and steam generators, in addition to the requirements of the Periodic Inspection standard. The purpose of the ISI program is to ensure continued reliable and safe performance of systems and components. The program draws on Ontario Hydro's operating experience, as well as other Canadian and international nuclear operating experience.

The data from the above inspection programs are collected and analyzed. In addition to influencing operating and maintenance activities, the data are used to confirm or refine reliability and performance predictions. Significant Event Reports (SER) are also generated at the stations which cover a wide range of events, including those related to component failures.

From the application of reliability and maintainability programs, a high level of confidence can be gained that components and systems are "ageing" as anticipated in the initial design specifications. This confidence is reinforced by continually reducing performance and ageing uncertainty through well defined programs of research, development, review and testing throughout the design life of each nuclear station.

16.6 Management of Reactor Ageing at Ontario Hydro

As has been mentioned previously, the ageing phenomena unique to nuclear power stations are those related to the neutron bombardment of in-core components and irradiation of components proximate to the reactor core.

For such components, there exists an extensive theoretical base of understanding which is continually being enhanced and reinforced by operational experience - a particular example being that of pressure tube ageing (see Section 9).

All components in a nuclear station, e.g., boilers, piping, valves, pumps, instrumentation and reactor components, etc. are designed and specified to achieve set lifetimes under the operating and environmental conditions expected to be encountered. Many systems and components are designed to achieve or exceed the full 40 year (economic) lifetime of the station and require simply routine monitoring and periodic maintenance activities. For components which are known to have shorter lifetimes, the design basis includes provision for ready replacement.

In-service inspection programs provide information on system and component performance to confirm the design basis. Any operating regimes, or environmental conditions, that are different from those included in the design basis are assessed for possible impact on predicted system and component lifetimes.

For systems and components that are specifically affected by ageing phenomena, (whether due to radiation or more conventional ageing conditions), world-wide experience and results from specific CANDU research and development programs provide the means to develop measures which could include the modification of operating regimes, environmental conditions, component material, or design.

Canada's demonstration power station, NPD, and the prototype which followed, Douglas Point, have been key sources of data related to CANDU station ageing phenomena. Ageing and operating experience from these stations complements specific research programs carried out in the laboratory. The experience in these stations showed that the thinning or cracking of heat exchanger U-tubes required attention in subsequent stations to reduce the effects of vibration induced fretting and mechanical wear. Other specific examples of ageing were degradation of turbine blades due to wet steam erosion, and the loss of flexibility of fuelling machine hoses due to radiation and thermal effects. In all instances ageing of components was detected either via detection schemes inherent to the particular system or during periodic inspections and testing. Corrective actions included repair or replacement with superior materials or changes in process system parameters or environmental conditions. This work is exhaustively documented through SERs and in-service and research reports for use by designers, operators at other stations, and manufacturers.

Ontario Hydro's CANDU reactors have demonstrated the capability to accommodate component failure while ensuring minimal (or zero) public risk. Nuclear station components will be continually monitored for ageing related phenomena and appropriate actions taken to ensure safe and reliable performance.

Costs of Nuclear Safety

SECTION 17: COSTS OF NUCLEAR SAFETY

17.1 INTRODUCTION

As explained in previous sections, nuclear safety considerations are included in all phases of the life cycle of the plant and in the design and operation of a large number of systems. The total costs related to nuclear safety provisions comprise the following major components:

(a) quality process systems and structures designed and installed to comply with stringent code requirements;

(b) design, engineering and installation of special safety systems including modifications (retrofits) during construction and operational phase;

(c) safety analyses, licensing and safety research and development.

(d) operator training.

17.1 Nuclear Quality Process System Costs in Perspective

A direct comparison of costs attributable to nuclear safety for the process system is not feasible since the requirements for a high quality process system are integrated into the safety approach which, at the first level of defence, requires prevention of failures. However, the contributory role of nuclear safety requirements can be inferred from a comparison of cost data with an alternative source of generation.

The nuclear process systems, such as the heat transport system, are designed to codes and standards which require the highest in material, design, fabrication and installation practices. These standards, developed for the nuclear industry, reflect a higher level of quality than is applied to conventional piping systems. A comparison of the cost data with a coal plant, therefore, provides some perspective on the overall capital costs and its influence on the total unit energy cost.

Table 17-1 provides a typical cost comparison between the Ontario Hydro Pickering Nuclear Generating Station (NGS) A and Lambton Thermal Generating Station (TGS) (Reference 1). The Pickering NGS A is comprised of four 515 MWe(net) nuclear units of the CANDU-PHW type. The Lambton TGS is comprised of four 495 MWe(net) units which burn coal. Both stations were built at the same time, both are of modern design and both stations are operational with good performance records. Table 17-2 provides a similar comparison between Bruce NGS A and Nanticoke TGS. The Pickering NGS A and the Lambton TGS were built in the late 1960s and were placed into service in the early 1970s whereas Bruce NGS-A came into service between 1977 and 1979. Table 17-1 and 17-2, therefore, not only provide a comparison between coal and nuclear but also a trend in time which indicate higher costs on Bruce NGS A resulting from the high inflationary period of the 1970s. It can be seen from Tables 17-1 and

17-2 that the Capital Cost for coal-fired stations is much lower (by a factor of 2.8) than the nuclear capital cost. The coal-fired OM&A Cost is lower by a factor of approximately 2.4 to 2.2 compared to the Nuclear OM&A cost.

Direct coal cost comparisons to the newer Pickering NGS-B and Bruce NGS-B nuclear stations are not possible since Ontario Hydro has not built any major new coal-fired stations after Nanticoke TGS. Cost comparisons with comparable new coal-fired stations built with to-day's scrubber technology, shows a much smaller difference in Specific Capital Costs (Reference 1). The specific capital cost ratio for Pickering NGS B ($1871/kWe) with its comparable wet-scrubber coal-fired station ($1271/kWe) is 1.47. Similarly, for Bruce NGS B ($1822/kWe) and its comparable wet-scrubber coal station ($1254/kWe), the ratio is 1.45.

Comparisons have also been made of the project cost of Ontario Hydro nuclear generating stations with nuclear generating stations in the United States (Reference 2). Figure 17-1 shows these comparisons on the basis of dollars per kilowatt (constant December 1986 Canadian dollars).

17.2 Special Safety Systems

The design approach requires that a second and third level of defence be provided to prevent releases of radioactive materials for expected failures and limit releases for postulated failures, respectively. The shutdown systems, the Emergency Coolant injection system and containment constitute the special safety systems. The primary role of these systems is to provide a safety function, and, therefore they are independent of the process systems.

The total costs associated with design and installation of these special safety systems are summarized in Table 17-3 for Pickering NGS B and Bruce NGS B and Darlington NGS A. The total costs documented in Table 17-3 provide an order of magnitude estimate to illustrate the level of expenditure on these systems. The costs quoted are the original costs incurred (actual costs in $ of the year). For example, the in-service dates for Units 1 to 4 for the three projects are as follows: Pickering NGS B (Mar 83 - Feb 86), Bruce NGS B (Mar 85 - May 87), Darlington NGS A (1988-1992) and, therefore, the varying impact of different inflationary periods must be accounted for before a direct comparison can be made of the costs of these systems amongsts the stations. It must also be noted that the costs for Darlington are based on an estimate to completion.

TABLE 17-1

Pickering NGS-A (Units 3 & 4)/Lambton TGS
Cost Comparison - 1985 (Reference 1)

Pickering NGS-A Units 3 & 4 and Lambton TGS Net Capacity Factor: 68.4%

	Total Unit Energy Cost (TUEC) [$/MW.he (net)]	
	Pickering NGS-A Units 3 and 4 (2 unit-years)	Lambton TGS (4 unit-years)
Interest, Depreciation and Decommissioning	12.07	2.19
Operation, Maintenance and Administration	6.16	2.52
Fuelling	4.20	23.32
Heavy Water Upkeep	0.89	-
Total Unit Energy Cost (Net)	23.32	28.03
1985 Net Energy Output (TWhe)	6.2	11.9

Station Data

	Pickering NGS-A	Lambton TGS
Capacity (Maximum Continuous Rating) MWe net	4 x 515	4 x 495
In-Service	1971 - 1973	1969 - 1970
Original Capital Cost (M$ Canadian escalated)	746.5	257.0
Specific Capital Cost ($/kWe)	362.4	129.8
Economic Lifetime (years)	40	35
Depreciation Method	Straight Line	Straight Line
Interest Rate (%)	13.0	13.0

TABLE 17-2

Bruce NGS-A/Nanticoke TGS Cost Comparison - 1985 (Reference 1)

Bruce NGS-A and Nanticoke TGS Net Station Capacity Factor: 81.8%

	Total Unit Energy Cost (TUEC) [$/MW.he (net)]	
	Bruce NGS-A (4 unit-years)	Nanticoke TGS (8 unit-years)
Interest, Depreciation and Decommissioning	13.76	3.77
Operation, Maintenance and Administration	3.99	1.79
Fuelling	5.10	26.02
Heavy Water Upkeep	0.35	-
Total Unit Energy Cost (Net)	23.20	31.58
1985 Net Energy Output (TWhe)	22.3	28.5

Station Data

	Bruce NGS-A	Nanticoke TGS
Capacity (Maximum Continuous Rating) MWe net	4 x 809	8 x 497
In-Service	1977 - 1979	1973 - 1978
Original Capital Cost (M$ Canadian escalated)	1 961.1	872.9
Specific Capital Cost ($/kWe)	606.0	219.5
Economic Lifetime (years)	40	35
Depreciation Method	Straight Line	Straight Line
Interest Rate (%)	13.0	13.0

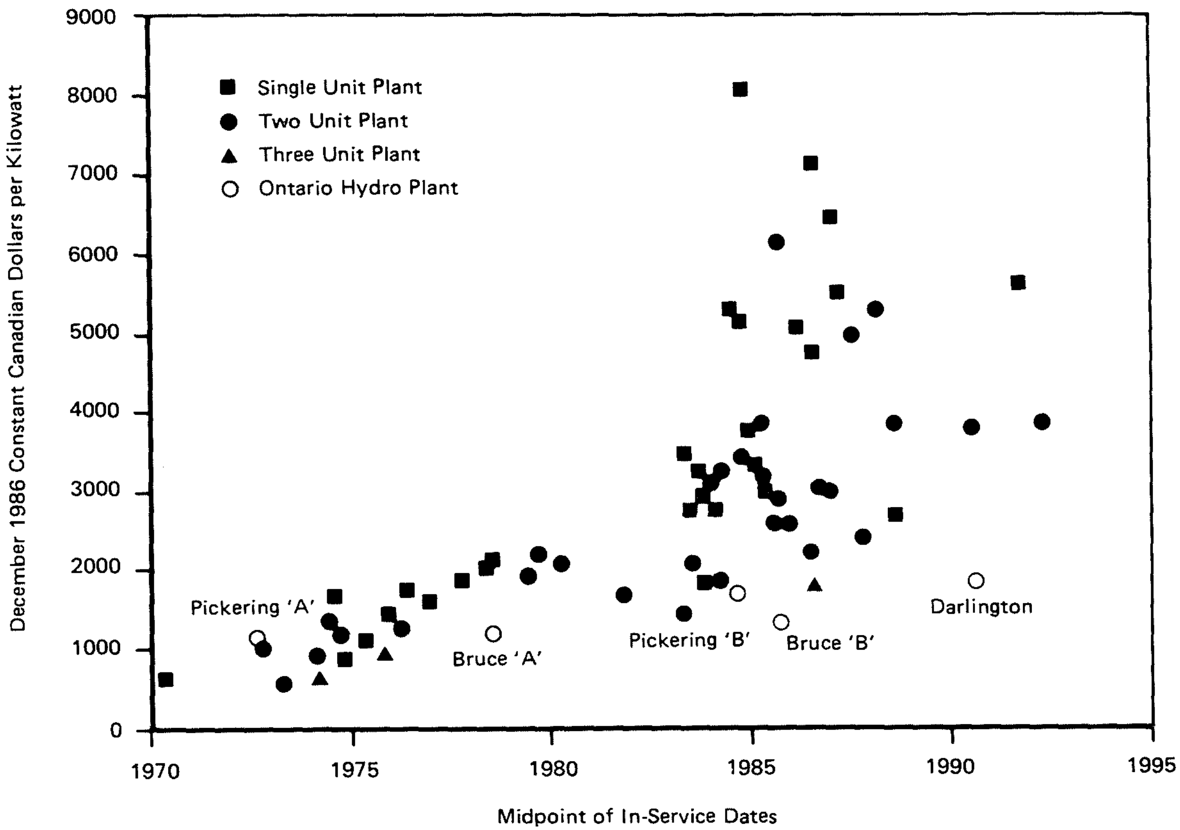


FIGURE 17-1
Electric Utility Cost Group Data: Nuclear Plants
Graphic Comparison of Total Project Cost per Kilowatt

17.3 Retrofits

Several design modifications, or retrofits, have been implemented at operating nuclear generating stations. The reasons for the changes and the associated costs are described below. The documented costs in this section pertain specifically to design, engineering, materials and installation costs for these modifications. Significant additional costs are incurred which result from the cost of replacement energy when the units are not available. These additional costs for replacement energy are not included because they vary widely for each specific change and there can be many reasons for extension of maintenance outages. However, to provide a measure for these costs, extension of outage for one unit at Bruce NGS-B for one week would result in additional cost of approximately $3 million.

17.3.1 Bruce NGS A

17.3.1.1 Instrumented Pressure Relief Valves

On Bruce NGS A, the containment system pressure relief valves that sit between the pressure relief duct and the vacuum building (as described in Section 3) comprise sixteen main pressure relief valves, four auxiliary pressure relief valves and two reverse flow valves. The main pressure relief valves are self-actuating and thus require some pressure in the pressure relief duct to operate them. The auxiliary pressure relief valves can be opened remotely by control systems at any pressure conditions, but are of limited capacity. To add diversity to the auxiliary pressure relief valves and to provide added capacity to cater to some specific low probability events, it was decided to add controls to four main pressure relief valves that would allow them also to operate at any duct pressure as well as in their self-actuating mode. The cost of this modification was $3 million.

17.3.1.2 Powerhouse Environment

In the event of failures in the steam or feedwater systems, the powerhouse will become hot and moist. To improve the confidence in the proper operation of equipment located in this environment that is credited with operation following such an event, a number of modifications have been undertaken. These modifications include:

(a) Provision of an emergency venting system which will reduce the temperature, pressure, and duration of any resultant powerhouse environmental transient.

(b) Provision of a specially environmentally qualified power supply and distribution system to ensure electrical supplies to essential equipment.

(c) Reinforcement and steam-proofing of essential areas and rooms such as the main control room and instrument rooms.

TABLE 17-3

Cost* of Special Safety Systems

Special Safety System	Pickering NGS B (k$)	Bruce NGS B (k$)	Darlington NGS A+ (k$)
Shutdown System 1 (SDS1)	24,000	45,000	100,000
Shutdown System 2 (SDS2)	28,000	44,000	90,000
Emergency Coolant Injection (ECI)	62,000	74,000	80,000
Containment	308,000	466,000	1,500,000

*Costs in Table 17-3 are the original capital costs incurred (i.e. actual costs in $ of the year).

+The costs for Darlington NGS A are based on an estimate to completion.

TABLE 17-4

Bruce NGS B
Licensing Related Design Changes

Change	Cost (k$)
PHT Pump Trip on LOCA	570
PHT Pump Low Speed Operation	27,000
ECI Seismic Qualification	770
Modifications for Harsh Powerhouse Environment	1,900
Modifications to IPRVs, APRVs	1,050
Moderator Seismic Qualification	2,100
Boiler Feedline Low Pressure Trips	1,100
High Pressure ECI	74,000
Emergency Filtered Air Discharge	4,400

(d) Qualified ventilation of essential areas.

(e) Additional pumps added to the Emergency boiler cooling system for improved diversity and reliability.

Total costs for these modifications is $12.6 million.

17.3.1.3 Emergency Filtered Air Discharge System

The original design of the Bruce NGS A credited the use of the pressure balance exhaust system to vent the containment system once it had returned to atmospheric pressure (see Section 3 for a description of the containment system operation). In order to:

(a) Improve reliability of this function;

(b) Improve capability of this function;

(c) Incorporate monitoring provisions;

it was decided to upgrade this venting function with the addition of a centralized Emergency Filtered Air Discharge system. Costs of this additional system are $5 million.

17.3.1.4 High Pressure Emergency Coolant Injection (HPECI)

During the final licensing stages of Bruce NGS A, it became apparent that there was a range of reactor header break sizes for which the gravity injection emergency coolant injection system could not be shown to meet the design target of preventing significant fuel sheath failures. The existing system was still able to meet the criteria for public dose under which it was to be licensed. In order to obtain the full power license, Ontario Hydro agreed to undertake a study to determine the benefits of a high pressure ECI system. The study showed that there would be benefits and Ontario Hydro committed its installation. The AECB issued an operating licence for Bruce NGS A but limited the power at which it was licenced to operate until a HPECI system was installed. Cost of this addition is $104 million.

17.3.2 Bruce NGS B

Bruce NGS B was intended to be a repeat of the Bruce NGS A design to the greatest extent practicable with modifications allowed only if:

(a) They were required for licensing reasons

(b) They would result in significant cost savings

(c) Equipment was no longer manufactured.

A number of modifications were made that were based on changing licensing requirements that were not included in the original design of Bruce NGS B but were installed or committed at the time of its operating license. A number of these modifications with costs in excess of 0.5 million are listed in Table 17-4.

17.3.3 Pickering NGS

17.3.3.1 High Pressure Emergency Coolant Injection (HPECI)

The original design of the Pickering A emergency coolant injection system was to inject moderator D_2O into the heat transport system when a loss of coolant accident was detected. When the supply of moderator was used up, then coolant and moderator would be recovered from sumps where it would collect by gravity and be re-injected by the moderator pumps. During operation of the plant, it proved difficult to demonstrate that the system met its reliability target. It was determined that system modifications would be required to meet these targets. Because of the Bruce NGS A licensing experience, the fact that a high pressure system had already been installed on Pickering B and a desire to be able to confidently predict the effectiveness of the ECI for small breaks, it was decided to extend the high pressure injection system that existed on Pickering B so that it could also inject into Pickering A. This high pressure source would allow operation of heat transport pumps which gives good cooling of the fuel throughout the small PHT break transient. The modification allowed increased reliability of the system by duplication of some of the injection valves, increased effectiveness by increasing the injection pressure, and increased confidence that there would be sufficient water in the recovery sumps by using an additional source of injection water. The cost of this modification was $90 million.

17.3.3.2 Powerhouse Environment

Similar to situation described in Section 17.3.1.2 for Bruce NGS A, it has been decided to undertake modifications in the Pickering NGS A powerhouse to ensure proper operation of credited equipment following a steam line or feedwater line failure. These modifications are estimated to cost $20 million.

17.3.3.3 Rupture Panels

In the Pickering NGS style of containment, there exists an atmospheric barrier between the reactor buildings and the pressure relief duct. In Pickering NGS B, this barrier is normally leak-tight, but when the reactor building exceeds a certain (fairly low) pressure, a number of panels will be opened by the pressure and provide a connection between the Reactor Building and the pressure relief duct. In Pickering NGS A the barrier consisted of an array of normally closed louvers, that would open on an overpressure, and a light membrane of aluminum foil.

In the event of a loss of coolant accident in some other unit, the atmospheric barrier in the Pickering NGS A units could not confidently be demonstrated to remain intact. Thus instrument air from these non-accident units would contribute to the repressurization of containment during its subatmospheric hold-up time and would also contribute to the air that would have to be vented once the containment returned to atmospheric pressure.

Ontario Hydro made strong representations to the AECB that further modifications were not justified.

Even though the predicted doses to the population in the event of a serious accident were already well below the published AECB licensing criteria and the radiation exposure to construction workers involved in a modification would be greater than the predicted dose saving to members of the public in the event of an accident, the AECB directed that modifications be implemented to double repressurization time. From among the options studied, Ontario Hydro chose to modify the atmospheric barriers in Pickering A to perform in a fashion similar to those installed in Pickering NGS B. The modification will extend the predicted repressurization time from 23 hours to 48 hours and will allow more time for off-site emergency planning activities to be implemented if such were necessary.

The cost of this modification is $6.5 million.

17.3.3.4 Filtered Air Discharge System Premonitoring

In the operation of the Containment System, following a subatmospheric holdup period, the containment is vented through the filtered air discharge system. The original design of this system at Pickering NGS had provision for monitoring the amount of radioactivity in the air that was being discharged from the containment. It was decided to improve the design to bring it in compliance with more modern standards so that the discharge could be monitored before release would commence. This is to be implemented with a recirculation line that will allow the discharge to be directed back into containment. Once it is confirmed that the filter system is operating correctly and releases could be within acceptable levels, then the discharge can be directed to atmosphere. The modification also allows "banking" of vacuum within the vacuum building so that releases can be suspended during periods of poor atmospheric dispersion.

The cost of this modification is $2.0 million.

17.3.3.5 Additional Shut-off Rods

During the retubing of the Pickering NGS Unit 1 and Unit 2 reactors, it was decided to augment the capability of the eleven shut-off rods by adding an additional ten shut-off rods by converting some adjuster rod sites to shut-off rod sites. This addition significantly enhances the depth and rate of the system, since the new rods and mechanisms are of the newer, spring assisted Pickering NGS B design.

The changes will also be added to Units 3 and 4 during the addition of high pressure emergency coolant injection during outages in 1988 and 1989.

The cost of this modification is $4.5 million.

17.3.4 All Stations

Following the experience of the Three Mile Island accident, as described in more detail in Section 10, it became necessary to add hydrogen mitigation systems to all of Ontario Hydro's reactor containments.

Costs of these additions are listed below in Table 17-5.

TABLE 17-5

Costs of Containment Hydrogen Mitigation Systems in Ontario Hydro Stations

Station	Cost (k$)
Pickering NGS A	400
Pickering NGS B	400
Bruce NGS A	1,100-1,300 (estimate)*
Bruce NGS B	600

*Installation not completed.

17.4 Nuclear Safety Assessment, Research and Development and Licensing

Ontario Hydro has developed a strong in-house capability in the area of nuclear safety analyses and licensing support for pre-operational and operating nuclear facilities. This knowledge base is, in addition, complemented by Ontario Hydro's active participation in large scale international Research and Development Programs.

There are several departments within Ontario Hydro which contribute to the process which result in a formal documentation of the safety report and operating policies and principles. The major departments involved in the process are Nuclear Studies & Safety Department (NSSD) in D&D-Gen, Radioactivity Management & Environmental Protection Department (RMEP) in TTSD and members of the technical and operating units at each of the stations, so it is difficult to give a definitive total for how much is spent on safety each year. Since safety is an integral part of the operation of a nuclear station, operating staff do not differentiate between normal operating tasks and safety related activities. Costs associated with the head office support groups, NSSD and RMEP can be determined. RMEP currently has a staff of 55 people devoted to operational safety support with an annual budget of approximately $5,000,000. NSSD costs for the last four years are shown in Table 17-6.

The NSSD figures have been broken down into three broad categories. The capital portion represents the cost associated with design assist analyses, safety report preparation, safety assessments and risk assessment for new plants. Ontario Hydro's in-house costs have increased over the years as they took over more responsibility for these studies from AECL.

The safety analyses and assessments carried out for Bruce NGS B by NSSD staff between January 1982 and December 1984 totalled 8.8 M$. The costs associated with Darlington during 1986 alone, were 8.3 M$. This increased cost for Darlington is attributable to two factors - increased scope and depth of analyses called for under trial use of C-6 licensing requirements and the Probabalistic Risk Assessment study. From the start of the Darlington project in 1978, NSSD has spent a total of 46 M$ on safety related studies.

The operational station costs shown in Table 17-6 result from assistance provided to the station in dealing with AECB licensing issues, safety related design problems and safety report updates. These costs include safety analyses for the restart of Units 1 and 2 at Pickering NGS A and analyses associated with the major design retrofit discussed in Section 17.3.

In order to confidently support the claims made in the safety reports, Ontario Hydro actively pursues a safety R&D program. The majority of the program costs are shared with AECL, Hydro Quebec and New Brunswick Electric Power Commission, the utilities and AECL sharing the costs on 50-50 basis under the COG program.

In addition to supporting R&D in Canada, Ontario Hydro participates in large scale international experiments. Over the last few years these have included jet blowdown and fission product transport experiments at Marviken in Sweden, fission product release and transport studies at Argonne National Laboratory, the Light Water Aerosole Experiments at Hanford and the large scale hydrogen tests in Nevada. While these were all large experiments, each involving ten's of millions of dollars, Ontario Hydro's relatively small share of these programs gained full access to the results for a total expenditure of 3.6 M$. One fully integrated CANDU specific fuel experiment was undertaken at the Power Burst Facility in Idaho at a cost of approximately 5 M$, shared equally between AECL and Ontario Hydro.

Safety studies are an evolving process. Limitations uncovered in models, new experimental information, and increased depth and scope of safety studies requires that methods used for analyses be continually updated. Ontario Hydro has, for example, devoted significant efforts toward the development of a two fluid model to better account for the separation of steam and water under accident conditions, and on better methods to provide realistic assessments of ECI system performance under certain large break loss-of-coolant scenarios. While these are classed as "methods development", they are in essence theoretical research programs requiring a high degree of skill and specialization with substantial associated costs. Currently NSSD is spending between 3 to 4 M$ annually on such studies.

17.5 Nuclear Training

The costs of authorization training for nuclear shift supervisors and unit first operators are documented in Table 17-7. The costs shown are the actual costs for 1984, 1985 and 1986 and projected costs for 1987 and 1988. All costs are in 1987 dollars.

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TABLE 17-6

Nuclear Studies and Safety Department Costs

	1984 (k$)	1985 (k$)	1986 (k$)	1987 (k$)	1988 (k$)
Capital Projects	6,967	5,345	9,054	5,594	5,115
Operational Station	3,184	5,600	3,255	4,189	4,702
Methods Development	2,800	2,965	3,404	3,184	4,716
Research & Development	4,941	6,244	6,470	7,264	7,608

TABLE 17-7

Nuclear Training Costs

	1984 (k$)	1985 (k$)	1986 (k$)	1987 (k$)	1988 (k$)
Training Delivery	1,788	1,577	1,799	1,828	1,873
Training Development	632	1,514	1,466	1,530	1,303
Simulator Maintenance	3,115	3,063	3,705	4,055	4,825
Simulator Depreciation	1,419	1,732	2,242	2,754	4,094
Trainee Costs	9,244	10,878	11,811	9,161	6,521
TOTAL	16,198	18,764	21,023	19,328	18,616

Enriched Uranium Fuel in CANDU Reactors

18.0 ENRICHED URANIUM FUEL IN CANDU REACTORS

One of the distinguishing features of the CANDU design is the use of natural uranium fuel (0.71 percent uranium-235) and heavy water moderator and coolant, arranged in an optimum array. In combination with low-neutron-absorbing in-core components made from zirconium alloys, this choice of materials results in a reactor design which is highly efficient in terms of uranium utilization. The result is that CANDU has the lowest fuelling cost of all reactor designs in commercial use today. In addition, this configuration eliminates any concern about possible re-establishment of a critical mass and production of fission energy following accidents or severe damage to the core in which the fuel is brought into closer relationship. This is one of the inherent safety features of the CANDU system.

It has long been recognized, however, that alternative fuels are available or could be developed which would reduce uranium consumption in CANDU reactors even further. By requiring less uranium to produce the same quantity of energy, alternative fuels have the potential to reduce fuelling costs and reduce the quantity of irradiated fuel bundles that must be stored and possibly transported for disposal.

Ontario Hydro has periodically reviewed international developments in nuclear fuel cycle technologies to determine if introducing an alternative fuel cycle into its nuclear generating system would result in any change to the inherent safety feature of natural uranium fuelling, or any net benefit to its customers. The simplest and probably the most economic alternative to fuelling with natural uranium, is to introduce a small amount of uranium-235 enrichment uniformly in the fuel bundles with no other changes in the fuel or reactor design.

Enrichment levels equal to that used in light water reactors throughout the world (about 3 percent uranium-235) are much too high for economic and practical application in the neutron efficient CANDU system. However, preliminary studies have shown that a very small increase in fissile content of the uranium (from 0.71 percent to 0.9 percent uranium-235) uniformly in the fuel in a CANDU would result in economic savings with no change in the inherent safety feature and no re-establishment of a critical mass following a postulated accident which causes a rearrangement of the fuel geometry in the core.

The savings in fuelling cost compared to natural uranium fuelling for this very small change in fuel enrichment are shown to be about 25 percent, and the reduction in irradiated fuel which must be stored and eventually disposed of is about 50 percent. Before these potential benefits could be realized, further studies and a demonstration of the adequacy of fuel performance under the extended burnup would be required. The fuel cladding must remain intact under irradiation in the reactor for about twice as long with this level of enrichment compared to natural uranium fuel.

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Concluding Remarks

ONTARIO NUCLEAR SAFETY REVIEW

CONCLUDING REMARKS

This report is of necessity a very condensed summary of a large volume of documented information related to nuclear safety and safety related issues, almost all of which is already in the public domain. Extensive lists of references and associated documents are included for those who may want more information on any of the subjects covered in this report.

Ontario Hydro is responsible for the safety of its nuclear stations: for the accuracy and relevance of the safety analysis, for the integrity of the design and construction, for the thoroughness of the training and competence of its operators and for the completeness and effectiveness of its operating and maintenance procedures and practices. While the AECB ensures that safety regulations are complied with, the utility must demonstrate to the regulatory body, and to the public, that it is competent and capable of operating nuclear stations safely.

This report conveys our intention that nuclear safety in Ontario Hydro is of highest importance and that dedicated attention of management and workers alike be given to the achievement of an excellent safety record. The management of nuclear safety in Ontario Hydro begins with well understood corporate goals, objectives and policies, and the clear assignment of responsibilities to well trained, competent people. These individuals have the relevant experience and the right information and equipment to plan, direct and control tasks, and to verify and monitor results, and to identify and remedy deficiencies. Reliable and safe nuclear power exists in Ontario because of the sincere attention of management and workers to safety.

A prime cause of the Chernobyl and Three Mile Island accidents was a breakdown in operational procedures and human factors. In both of these accidents, the operators did not seem to understand what was happening within the nuclear station from information available to them in the control room and this contributed to the severity of the accident. They did not take timely or appropriate action to prevent these accidents from occurring.

On the contrary, the pressure tube failure at Pickering Unit 2 on August 1, 1983 was understood almost immediately by the operators and they took the correct steps to shut the unit down in an orderly manner and to cool and depressurize the reactor systems. We relate this success to well designed control room information systems, and good understanding of fundamentals by the operator(s). This is reinforced by extensive training on the nuclear plant systems and practice on training simulators on the behaviour of transients and upsets. Nevertheless there is no room for complacency.

There is increasing awareness world wide of the need to consider the human as a major system component in nuclear power, taking advantage of and making allowances for human capabilities and limitations. Although humans are enormously adaptable and can meet extreme demands by pushing the limits of their performance, prudence would dictate it is better and more efficient to concentrate their efforts on functions which reflect their normal capabilities and which are not compromised by their natural limitations. Increasingly, in the design of nuclear plant control and instrumentation systems, and in training operators in Ontario Hydro, the well-being, capabilities and limitations of human beings are being taken into account by a systematic application of human factors engineering.

This report describes the series of barriers between the radioactive material in the fuel and the public environment. These barriers shield against radiation and contain radioactive materials. The report also describes the stringent quality control and numerous technical measures taken in Ontario's CANDU nuclear stations that make the likelihood of malfunctions, which could lead to a threat to these multiple barriers, very small. Even so, equipment malfunctions and human errors must be expected and safety designs are based on the postulation that there will be no significant release of radioactive materials to the environment for all credible combinations of malfunctions. The high quality process system design and inherent safety characteristics of the CANDU reactor, backed up by reliable and powerful safety and containment systems, provides the needed defence-in-depth for safe nuclear power stations. This defence-in-depth is of vital importance in protecting the public against any conceivable combination of equipment failures, or human errors, or both.

This defence-in-depth protection for the public is a feature of all Ontario Hydro nuclear stations. As each successive nuclear station in Ontario was committed, safety and safety related systems were updated, based on the latest information and technology or due to changed regulations. Some of these improvements were, or are being, backfitted to operating nuclear stations, specifically where significant improvement in safety might result. Others were not backfitted, where existing systems had demonstrable and acceptable safety performance and where even extensive backfitting would not improve significantly the already very high level of safety performance. In these cases, the performance of the safety systems in these operating stations has been demonstrated by continual testing for many years. A case in point is the "single" shut down system in Pickering A, which consists of both gravity drop shut-off rods and moderator dump, both triggered by the same triplicated set of trip signals. This arrangement has speed and depth and diversity in the shut down mechanism, and has a very high level of reliability from tests since 1971. We have, however, recently increased the number of shut-off rods in Pickering Units 1 and 2, and plan to make this change to Units 3 and 4 in the near future, to bring this fast shut-off capability up to that in the most recent plants.

Reliable and safe nuclear station performance are both public and utility requirements of vital importance. Ontario Hydro is committed to these mutually supportive requirements.

Appendix A
Bibliography for CANDU Reactor Design and Accident Analysis

Appendix A

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Appendix B
Documents Submitted by Ontario Hydro to the Ontario Nuclear Safety Review

APPENDIX B

This appendix comprises a list of documentation submitted by Ontario Hydro to the Ontario Nuclear Safety Review.

ONTARIO NUCLEAR SAFETY REVIEW

Information Provided by Ontario Hydro

Date	Item Provided
Feb 19/87	Final Report, Selection Committee on Hydro Affairs on "The Safety of Ontario's Nuclear Reactors", June 1980. (2)
Feb 19/87	Presentation to Selection Committee on Energy by J. McCredie, September 1985.
Feb 19/87	Hydro telephone directories (2)
Mar 16/87	Excerpt, Goldfarb study of public attitudes to nuclear power, July '86.
Mar 17/87	NIRC & NEPC terms of reference.
Mar 17/87	NIRC Annual reports, 1983 to 1985.
Mar 25/87	Nuclear brochures, pamphlets and cutaway drawings.
Apr 9/87	Videotape: "Nuclear Power: In France It Works", 1987 CBS News Feature.
Apr 9/87	Toronto Star clipping, "Soviets look to Hydro for nuclear plant safety", November 15/86, R. Spiers.
Apr 15/87	Aerial photos, Bruce, Pickering & Darlington stations.
Apr 15/87	Bruce GS "A" Safety Report, 3 volumes.
Apr 15/87	Darlington GS Safety Report, 3 volumes.
Apr 15/87	Pickering GS "A" Safety Report, 1978, 2 volumes.
Apr 15/87	Pickering GS A Safety Report, 1987 (draft), 1 volume.
Apr 15/87	Report #85362, October '85, "Pickering NGS A Units 1 and 2 Restart Analysis".

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Date	Item Provided
Apr 15/87	Report #81157, April '81, "Pickering NGS A, Assessment of Large-Break Loss-of-Coolant Accidents".
Apr 15/87	Pickering GS B Safety Report, 1985, 2 volumes.
Apr 15/87	"Summary Report on the Post-Accident Review Meeting on the Chernobyl Accident", IAEA, Vienna, 1986.
Apr 15/87	"Pickering NGS A, An Assessment of Shutdown System Reliability...", Report #82497, October, 1982.
Apr 27/87	Copies of overhead slides illustrating system operation, Richview Control Centre.
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Apr 27/87	Reactor Building Dome", Pickering A Design Manual #44-21149, February 1970.
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Apr 27/87	"Reactor Building - General", Pickering A Design Manual #44-21000, December 1972.
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Apr 27/87	"Reactor Building Floor Loadings", Pickering B Design Manual #NK30-21040, August 1979.

Date	Item Provided
Apr 27/87	"Pickering NGS B: Post-LOCA Hydrogen Generation", Design & Development Division, Report #82240, April 1982.
Apr 27/87	"Pickering NGS B: Structural Integrity of Containment Following Pipe Breaks in the Steam and Feedwater Systems", Design & Development Report #83362, August 1983.
Apr 27/87	"Pickering NGS B: Containment Behaviour Following Loss-of-Coolant Accidents", Design & Development Report #82006, April 1982.
Apr 27/87	"Pickering NGS B: Extreme Containment Impairments", Design & Development Report #82007, March 1982.
Apr 27/87	"Reactor Buildings - Structures", Bruce A Design Manual #NK21-21100, January 1976.
Apr 27/87	"Shield Tank, Reactivity Mechanism Platform and Support Bearings", Bruce A Design Manual #NK21-31300, May 1973.
Apr 27/87	"Reactor Building", Bruce B Design Manual #NK29-21140, February 1982.
Apr 27/87	"Reactivity Mechanism Deck", Bruce B Design Manual DM-29-31360, November 1983.
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Date	Item Provided
Apr 27/87	"Containment", Darlington Nuclear Safety Design Guide DG-38-03650-6, February 1984.
Apr 27/87	"Reactor Building Structure", Darlington Design Manual #21100, January 1987.
Apr 27/87	"Reactivity Mechanisms Deck and Containment Seal", Darlington Design Manual #31710-1, September 1986.
May 5/87	List of unions involved in Hydro's nuclear design, construction and operation.
May 13/87	List of documents filed by Corporation. Rel. Hydro @ 1979 Select Committee on Hydro Affairs.
May 14/87	Nuclear station planned outage schedule, 1987/88, Rev. 87-04-10.
May 15/87	List of Bruce County officials.
May 20/87	Final report, AECL-Hydro D&D Gen. Joint Pressure Tube Project, re: Pickering GS tube failure, December 1984.
May 21/87	Information re: Bruce A Corporation. Rel. vacuum building valve failure during douse test.
May 21/87	Information re: Bruce GS A Unit 2 Channel N06 Failure.
May 26/87	Ontario Hydro 1986 Annual Report (2 copies).
June 9/87	INPO trip report re: technical exchange visit to Bruce NGS A.
June 12/87	Public Radiation Dose Calculation Guidelines for Hypothetical Nuclear Accidents, Draft CSA Standard N288.2.
June 12/87	Nuclear Emergency Response Strategy for On-Site Meteorological Monitoring, Report #OHNEP-1.
June 12/87	Preliminary Study, Dispersion Climatology, Bruce Area. Research Report #86-6-K.

Date	Item Provided
June 12/87	Dispersion Climatology, Pickering Area, Research Report #84-137-K.
June 12/87	Exposure Control Phase Procedures, Provincial Operations Centre, Technical Group.
June 12/87	Ontario Hydro Continuous Release Dispersion Model, Research Report #83-277-K.
June 12/87	Dispersion Modelling for Nuclear Emergency Response. Research Report #86-366-K.
June 16/87	Nuclear station planned outage schedule, 1987/88, Rev. 87-05-28.
June 16/87	"Pickering NGS A, An Assessment of Shutdown System Reliability...", Report #82497, October, 1982.
June 16/87	"Consequences of Pressure/Calandria Tube Failure ...During Full Power Operation, Report #84301, October '84.
June 16/87	"Pickering NGS A Units 1 & 2 Restart Analysis" Report #85362, October '85.
June 17/87	"Bruce NGS A Safety Report" Volume 1, 2 & 3, March '84
June 17/87	"Pickering NGS A Units 1 & 2 Restart Analysis" Report #85362, October '85.
June 24/87	"Pickering NGS A Units 1 & 2 Restart Analysis" Report #85362, October '85.
June 24/87	"Pickering NGS B Safety Report, Volume 2, February 1974, Revised January 1984.

Appendix C
Biological Effects of Exposure to Ionizing Radiation

APPENDIX C

BIOLOGICAL EFFECTS AND RISKS OF EXPOSURE TO IONIZING RADIATION

C.1.0 INTRODUCTION

The purpose of this Appendix is to summarize current knowledge with regard to the various adverse biological effects which can result to man following his exposure to ionizing radiation. In addition, the environmental consequences in terms of the effects on plants, trees and animals are also documented.

Broadly, the effects on man can be divided into two general classes (Reference 1). Non-stochastic effects, such as skin burns and the formation of cataracts, are those effects for which both the probability of occurrence and the severity of effect are a function of dose, and therefore for which a threshold of dose-response, which depends on the sensitivity of observation, may occur. In contrast, stochastic effects are those for which only the probability of occurrence, and not the severity, depends on the dose. While this definition does not, of itself, preclude a threshold, the uncertainty associated with the measurement of random events which occur with low probability means that a dose threshold is very difficult to demonstrate, and is rarely assumed. The principal stochastic effects are cancer and heritable (genetic) changes.

Within each class other distinctions can be made. In earlier days, radiation protection was directed, at avoiding the early effects of radiation damage, that is, those effects which become evident within hours to days after the exposure. The appearance of early effects requires that a significant dose be received acutely, and these tend to be the effects of most concern in accident situations. As the practice of radiation protection has matured, however, so that large acute exposures are uncommon in routine operations, concern has shifted toward control of those late effects which become evident only months to many years after exposure. Radiation carcinogenesis is such an effect, because the developments of all cancers are complex processes involving a series of steps which occur over a substantial fraction of an individual's lifetime. Similarly the expression of genetic damage must await the production of one or more generations of offspring. Because these two principal late effects are stochastic disorders which appear to depend only on the total absorbed dose, and not on its distribution in time, late effects are those primarily associated with chronic low-level exposure.

The aims of radiation protection, as identified by the ICRP (Reference 1), are: (a) to prevent all detrimental non-stochastic effects; and (b) to limit the probability of stochastic effects to levels as low as reasonably achievable. A more complete discussion of the distinctions described above, as a basis for this philosophy, is found in their Publication 26 (1977) (Reference 1).

C.2.0 NON-STOCHASTIC EFFECTS

Non-stochastic effects are those for which the severity, as well as the probability, depends on dose, and for which thresholds below which changes of clinical concern do not occur, often can be demonstrated. In Publication 41 (Reference 2) the ICRP provides a thorough review of the non-stochastic effects which occur in man, based mainly on the large human exposures which occur in medical radiation therapy. UNSCEAR (1982) (Reference 3) provides a similar analysis, with more emphasis on the details of laboratory studies.

Non-stochastic effects are those types of damage that result from the collective injury of substantial numbers or proportions of cells in an organ or tissue, or which serve a unique function. The essential feature of most non-stochastic damage is the killing of cells, and therefore all the complex cellular responses to injury, such as repair and repopulation which determine the outcome. These responses can depend not just on dose, but on dose rate, and on a host of biochemical and physiological characteristics of the environment provided by the individual. For this reason variations in sensitivity with, for example, age and sex are likely to occur.

In the sections to follow the responses of the most important organs or tissues the body with respect to non-stochastic radiation injury are described. A summary of all radiation risks according to level of exposure is provided in Section C.4.0.

C.2.1 Immediate Effects of Large Doses

Blood cells are among the most radiosensitive in the human body and measurable decreases in the number of circulating lymphocytes can occur following acute whole body radiation exposures above about 50 mSv (References 12,13). Lymphocyte chromosome aberrations are also observable above about 50 mSv (Reference 14), this effect having been used to develop a dosimetric method which can provide quite reliable dose estimates from blood samples (Reference 15).

At doses up to about 2 Sv, blood system damage is reversible and a healthy person usually recovers. At doses considerably above 2 Sv however, blood and blood forming tissue damage becomes massive and the additional effect of damage to other sensitive tissues such as the lining of the stomach and intestines, leads to a rapidly increasing risk of death. At about 6 Sv, the risk of surviving 30 or more days is on average about 50 percent, (this dose is called the $LD_{50/30}$, with most deaths occurring one to three weeks after exposure (References 12,13,16). At doses above 10 Sv, death for all persons so exposed is inevitable within a few days. At doses above 100 Sv, death occurs within hours due to widespread destruction of cellular matter, especially that of the central nervous system. The precise value of the $LD_{50/30}$ varies accordingly to the individual and the circumstances. The highest values will apply to healthy young adults.

The nature of medical attention also plays a role and has increased the $LD_{50/30}$ value in recent years as experience has increased in the treatment of acute dose victims. Acute radiation injury at various doses is summarized in the following table which is adapted from (Reference 17).

TABLE C-1

Acute Radiation Injury as a Function of Dose (Reference 17)

Dose (Sv)	Character	Symptoms
<0.05	No measurable effect	None
0.05-0.5	Leukocyte chromosome effects	None
0.5-2	Hematopoetic	Most of these patients are asymptomatic A few may develop mild dizziness and nausea. Some decrease in white blood cells and blood platelets may occur.
2-4.5	Hematopoetic	These patients develop acute radiation syndrome in a mild form. After transient nausea and vomiting, laboratory and mild clinical evidence for mild hematopoetic derangement dominates.
4.5-6	Hematopoetic	A serious course occurs in these patients. Complications of hematologic malfunctions are severe and, in the upper part of the group,some evidence of gastrointestinal damage may also be present.
6-10	Gastrointestinal	An accelerated version of the acute radiation syndrome occurs. Complications of gastrointestinal injury dominate the clinical picture. The severity of hematopoetic disturbances are related to the length of survival time following exposure.
>10	Central Nervous System	Fulminating course with marked central nervous system impairment occurs in this group.

Some acute internal exposure situations can lead to damage to a specific organ if the radionuclides accumulate in those organs. Some radionuclides are intense beta ray emitters and when present outside the body within the range of the beta rays, can cause skin damage leading to skin burns. Extensive skin burns from this cause, plus high levels of whole body dose, led to the deaths of several fire fighters at the Chernobyl accident (Reference 16).

Large doses from radionuclides entering the body by inhalation or ingestion are possible in accident situations. Such internal exposures may be uniform to the body or may affect particular body organs. Inhalation of tritiated air moisture, or ingestion of tritiated food or water, leads to uniform irradiation of the soft tissues of the body. For extremely large intakes, the effects would be similar to those for whole-body gamma ray exposures. Since radioiodines are concentrated in the thyroid gland, exposures to radioiodines which give little whole body exposure can nevertheless cause large doses to the comparatively small thyroid gland and surrounding tissues. In extreme cases, the thyroid may be destroyed. Sections C.2.1.1 to C.2.1.8 provide further details on an organ specific basis.

C.2.1.1 Skin

The irradiation of skin causes a progressively serious acute response leading from transient erythema (reddening) through moist desquamation to necrosis. The severity of the response depends on the dose absorbed by the germinal cells of the basal epithelium, which lies some 100 µm below the surface of the skin. For this reason even relatively high doses from alpha and weak-beta emitters cause little skin damage, since their emissions do not penetrate to this depth.

Some degree of radiation damage to skin is readily repaired. Thus the threshold for induction of mild erythema is about 600-800 rads* in a single exposure, rising to more than 3000 rads if the exposure is prolonged over a period of several weeks. The thresholds for desquamation and necrosis are higher, in the latter case 3000 rads of acute exposure and 9000 rads if the exposure is prolonged. The sensitivity of hair follicles, however, is greater. An acute dose of 300-500 rads results in a temporary loss of hair, and this loss becomes permanent after an acute dose of 700 rads, although 5000-6000 rads if the exposure is prolonged.

In addition to these early responses, late development of fibrosis can occur following high (1000-3000 rads) doses to deeper regions of the dermis.

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* 100 rads is approximately equal to 1 Sv.

C.2.1.2 Hematopoetic System

The dividing cells of the hematopoetic (blood-forming) system are among the most radiosensitive in the body. Following an acute whole body exposure exceeding 100 rads, changes in the bone marrow and lymphoid follicles, and in the peripheral blood count, can be observed within hours. The maximum depression, however, occurs only after a period of roughly 2-5 weeks. The depression of leukocytes (white blood cells) leads to a marked weakening of the immune response, and of resistance to infection. At about the same time a decrease in blood platelets, elements necessary to prevent bleeding, will be observed. After whole body doses of several hundred rads, infection and hemorrhage are the main causes of death.

Although it is estimated that the acute dose necessary to cause death in 50 percent of exposed persons is ≈ 500 rads, this outcome depends strongly on the medical support available during this critical phase. It is believed that several of the survivors of the accident at Chernobyl received bone marrow doses exceeding 700-800 rads. However if the patient survives this relatively short critical period, recovery is essentially complete and no lasting non-stochastic effects are expected.

C.2.1.3 Gastrointestinal Tract

The sensitivity of the cells of the gastrointestinal tract is similar to those of the skin. Killing of sufficient numbers of germinal cells of the mucosal epithelium causes ulceration, and subsequent fatal dysentry, over periods of 3 to 10 days if a large part of the intestine is exposed acutely to doses in excess of 1000 rads. Irradiation of the stomach to doses in excess of 50 rads contributes to a general condition of shock whereby 2 to 5 hours after exposure the subject experiences nausea, vomiting and diarrhea. These symptoms, however, are transient.

C.2.1.4 Reproductive System

The germ cells of the testis and ovary are also highly radiosensitive. In the testis the primitive cells that differentiate into sperm are most radiosensitive, so that depression in the sperm count and a loss of fertility will be observed several weeks after exposures exceeding ≈ 150 rads. This loss is temporary and recovery will occur over a period of months. After doses of 300-500rads, however, sterility may be permanent.

In the human ovary, all cells in the oogonial stages progress to oocytes before birth. Since there are no proliferating stem cells in the adult, the supply of oocytes is progressively lost through aging and ovulation. In the adult the most radiosensitive of the various maturation stages is the mature oocyte. Acute exposure of both ovaries to doses in excess of 65-150 rads leads to prompt loss of fertility. At doses less than 200-300 rads the loss is temporary, and fertility is recovered. This threshold, however, depends strongly on age. A dose of 300 rad to the ovaries of a woman aged 40 would almost certainly cause permanent sterility.

Even more sensitive are the developing oocytes of the female foetus. In laboratory studies even fractionated exposures to a total of 200 rads caused severe damage to the developing ovaries of the dog and the monkey, and permanent sterility. No comparable human data is available.

C.2.1.5. Thyroid

One of the main concerns with regard to accidental releases from nuclear facilities are the exposures of the thyroid which could result from intakes of radioiodines. The adult thyroid, however is relatively radioresistent with respect to non-stochastic effects. The threshold for severe function damage to the normal adult thyroid is about 2500 to 3000 rads. On the other hand, this relative radioresistance is strongly age-dependent. Marked thyroid depression, with accompanying retardation of growth, has been observed in children under 10 years of age following thyroid doses of only 700-1400 rads, and sub-clinical changes in function have been observed in those receiving doses of only 400 rads. There is no evidence, however, for any effect on thyroid function following doses of only a few 10's of rads.

C.2.1.6 Eye

Doses to the anterior epithelial cells of the eye lead, over a period of months at high doses, or many years at lower doses, to opacities of the lens, or cataracts. Microscopic abnormalities in these cells become detectable within minutes of exposures greater than ≈ 100 rads, but these changes do not affect vision. The progression to clinically-significant opacities, if it occurs, takes much longer, and full expression requires doses in excess of ≈ 500 rads.

C.2.1.7 Central Nervous System

The central nervous system is relatively radiation resistant. In patients receiving large doses to the brain or spinal cord during radiation therapy, myelitis will develop over several years but only following exposures in excess of ≈ 3000 rads. Acute exposures greater than ≈ 5000 rad, however, will cause severe acute damage leading to death in 1 or 2 days, or less.

C.2.1.8 Developing Embryo and Foetus

The most sensitive tissue of the human body with respect to the induction of non-stochastic damage by ionizing radiation is the developing embryo or foetus. Because of the rapid cell proliferation and cell migration, and the coordination of structure and function that must occur over a period of only 40 weeks many opportunities for serious disruption of this exquisite pattern present themselves.

UNSCEAR (1986) (Reference 4) provides a comprehensive review of biological effects of pre-natal irradiation. Briefly the development of the human conceptus can be divided into three phases: the pre-implantation period lasting from fertilization until implantation of the embryo into the uterine wall; the phase of major organogenesis, which extends in man until about the 8th week after ovulation; and the phase of foetal development, continuing on until birth.

Much of the quantitative information concerning non-stochastic effects of pre-natal irradiation is derived from laboratory observations in other species, and quantitative extrapolation between species is not sound. The major effect of irradiation during the first phase is death of the conceptus (technically a stochastic effect), but those that survive appear unimpaired with respect to morphology, size, long-term survival and reproductive fitness (Reference 5). In humans the effect would simply be noted as a temporary failure to conceive.

In many teratological effects of the types observed in experimental animals irradiated during the second phase are uncommon. In 1977 UNSCEAR (Reference 6) estimated the incidence of malformed fetuses per unit absorbed dose of the order of 5×10^{-3} per rad. The background rate of congenital abnormalities in man, however, is typically 6 percent of live births. UNSCEAR also estimated, at that time, the incidence of severe mental retardation, the most serious abnormality found in the offspring of survivors of Hiroshima and Nagasaki, as 10^{-3} per rad. The main concern was that such damage could occur during the first two months, when the woman might not yet be aware of her pregnancy.

More recently this data, derived from the survivors of the atomic explosions in Japan who were irradiated in utero, has been reanalyzed (References 4, 7, 8). Three significant conclusions can be drawn:

(a) The period of maximum sensitivity for induction of severe mental retardation occurs from the 8th to the 15th week of pregnancy. No excess risk was observed for the period 0-7 weeks, or from the 26th week until term.

(b) The rate of increase of incidence of severe mental retardation is high, about 0.4 per 100 rads. The data, however, cannot exclude a threshold of up to 10 rads.

(c) A correlation was observed between deficits in intelligence test scores, which represent a much more subtle measure at mental damage, and dose during these same periods, but the form of the dose response relationship has not been established. If linear without threshold, the data suggest a loss of 0.2 test points per rad during the critical period.

In comparison with the thresholds associated with lasting non-stochastic effects in other organs and tissues, hundreds and thousands of rads, the developing embryo and foetus are uniquely sensitive, and suffer serious consequences after doses of only a few tens of rads, if exposure occurs during particular periods of development.

C.2.2 Environmental Consequences: Effects on Biota and Natural Communities

Large radiation doses can damage or kill plants, trees and animals. The $LD_{50/30}$ for animals varies according to species (References 13,18). Large species have generally lower $LD_{50/30}$ values. Vegetation displays a wide range of radiation sensitivity according to species. Insects are generally very tolerant to high doses (Reference 13). These levels of dose are summarized in Tables C-2 and C-3.

TABLE C-2

Radiation $LD_{50/30}$ Values for Various Species

Species	$LD_{50/30}$ (Sv)
Dog	2.65
Guinea pig	2.55
Mouse	9
Rat	9
Rabbit	8.4
Song sparrow	8
Sheep	1.55
Pig	1.95
Goldfish	23
Fruit fly	>640
Amoeba	1000

TABLE C-3

Effect of Large Radiation Doses on Natural Communities

Communities	Dose (in Sv) for Level of Damage		
	Minor	Intermediate	Severe
Coniferous forest	2	2-20	> 20
Deciduous forest	2	2-100	>100
Grassland	20	20-200	>200
Herbaceous annuals	40	40-700	>700

Trees and plants, animals, birds, fish in streams or ponds, farm or horticultural products down wind of a nuclear facility, would be subject to exposure from accidental emissions of radioactivitY and might also become contaminated. This contamination could lead to doses being received externally by biota over a long period following an accidental emission from the radionuclides deposited on the surfaces and deposition on ground surfaces. Further long term doses may result from uptake of certain radionuclides from the soil by plants, leading to incorporation of these radionuclides into animal food chains.

Soluble radionuclides deposited on the ground would eventually find their way into ground water and surface water, and be carried into streams, rivers and lakes. Here, uptakes and exposures to aquatic plants and fish would occur, with much of the dose being internal.

Direct radioactive emissions to water would give immersion doses to fish and aquatic plants, but the major long term doses would be from uptake of radionuclides through the food chain.

Some plants, animals and aquatic organisms have the ability to concentrate certain radionuclides, sometimes by large factors. In other cases, particular radionuclides are not taken up by plants and therefore do not enter the food chain.

Of those radionuclides which could be emitted in a major nuclear accident, strontium-90 and cesium-137 are the major long-term concern. These radionuclides have a relatively large fission yield, long half lives, (≈30 years) and are taken up by plants.

C.3.0 STOCHASTIC EFFECTS

The main concern regarding chronic, low-level exposures to ionizing radiation are stochastic effects, for which the probability of occurrence, but not the severity, depends on dose. Thus when large populations are irradiated some individuals, though few in number, will experience the full consequences associated with that effect.

The principal forms of stochastic effects associated with exposure to ionizing radiation are cancer induction and heritable (genetic) changes. In contrast to the mechanism of non-stochastic damage, both effects are thought to arise from changes in only one or a few cells, probably unrepaired changes to the DNA which are passed on during cell division. Because these changes are unrepaired, the effects of each increment of chronic exposure are cumulative, so that the outcome depends primarily on the total dose, and not on its distribution in time. Other agents, such as chemicals or ultraviolet light, appear to cause similar damage so that, in general, both the cancers and genetic changes associated with exposure to ionizing radiation cannot be separated from those occurring in its absence. In such circumstances it is difficult to demonstrate the form of the dose-response relationship at low doses, and in particular the existence of a threshold, and consequently a linear, no threshold response is commonly assumed for small doses. Whether this assumption can be maintained to the relatively high levels of exposure for which effect data has been recorded is the subject of continuing controversy (Reference 5). In general such an assumption is regarded as conservative (Reference 1).

C.3.1 Radiation Carcinogenesis

The chief somatic form of stochastic radiation damage is cancer induction. Clear evidence showing an increased incidence of human cancer following exposure to large doses of radiation is provided by studies of the survivors of the atomic bombings of Hiroshima end Nagasaki, and of patients receiving large doses of radiation as part of medical treatment. These and similar studies have been reviewed in detail by UNSCEAR (1986) (Reference 4). In addition a large body of supportive evidence derived from experimental studies in the laboratory has been compiled (see, for example, Reference 9 (1986)).

The main source of information concerning the induction of human cancer by radiation remains the studies of the Japanese survivors, because this population, many of whom received doses up to 100 rad and more, has been carefully followed for a period of greater than 40 years. Two conclusions can be drawn:

(a) Like cancers attributed to other causes, radiation induced cancers exhibit substantial latent periods between exposure and diagnosis, so that an excess incidence does not appear for a period of 10 years after exposures for many solid tumours, and remains elevated thereafter to at least 30 years. For leukemia these periods are somewhat shorter, 2 years and 20 years, respectively;

(b) The dependence of incidence on age appears consistent, for many forms of cancer, with a relative risk model, wherein the excess risk, for a given dose, increases together with spontaneous incidence, with age. These conclusions have been incorporated into the recommendations of a working group of the US National Institutes of Health on estimating attributable risk (Reference 10).

Subject to any changes which may result from an on-going review of the dosimetry for these Japanese survivors, the risk of fatal cancers recommended in 1977 by the ICRP (Reference 1), and shown in Table C-4 remain valid. These risks are intermediate among the estimates published by UNSCEAR (References 4, 6) and the BEIR Committee (Reference 5).

TABLE C-4

Organ or Tissue		Risk Per Rem
Red Bone Marrow		2.0×10^{-8}
Bone Surfaces		0.5×10^{-8}
Lung		2.0×10^{-8}
Thyroid		0.5×10^{-8}
Breast		2.5×10^{-8}
Other Tissues		5.0×10^{-8}
	TOTAL	12.5×10^{-8}

In addition, a similar number of non-fatal cancers are expected.

These risk estimates are derived for a population of workers, whereas the actual risks vary with age, and with sex. In Publication 26 (1977) (Reference 1) the ICRP examined these differences; the risk of fatal cancer per rem exposure is higher by a factor of about 5 for males age 20 compared to age 60. This increased risk is due, mainly, to the limited period of expression for radiation cancers remaining at age 60 in a normal lifetime. Similarly the risk of fatal cancer per rem exposure is about 50 percent greater on average for females, compared to males. This excess is due mainly to the high sensitivity to radiation of the female breast.

Publication 27 (Reference 11) also quotes a value, published by UNSCEAR (Reference 6), for the induction of leukemia in an irradiated foetus. This rate, 2.3×10^{-4} per rad, is substantially higher than that in adults, but while many associations between excess incidence of leukemia and in utero radiation exposure have been published, consistent evidence is lacking (Reference 4).

C.3.2 Genetic Changes Induced by Ionizing Radiation

The second principal stochastic effect of exposure to ionizing radiation is the genetic changes, or mutations, expressed in future generations resulting from irradiation of the germ cells of the testes or ovaries. In contrast to the understanding of radiogenic cancer in humans, knowledge of genetic changes is limited almost entirely to that derived from laboratory studies, usually of mice. UNSCEAR (1986) (Reference 4) provides a detailed summary of this situation. No increased incidence has been observed to date among the offspring of the survivors form Hiroshima and Nagasaki. From laboratory data, ICRP 91 concluded that the risk of serious hereditary defects expressed in the first two generations is about 4×10^{-5} per rem, and the risk to all future generations is about twice that value (8×10^{-5} per rem). These estimates are similar to the current values provided by UNSCEAR (References 3, 4). In comparison roughly 10 percent of all live-born children carry some type of genetic or partially genetic defect of different grades of severity (Reference 3).

The sensitivity of female germ cells is probably much lower than that of male germ cells (Reference 3) and given their ore limited period of risk, females probably contribute much less than males to the burden of radiation-induced genetic changes.

C.4.0 SUMMARY OF RADIATION RISKS

The preceding sections have summarized by target organ and type of damage the risk of exposure to ionizing radiation. For non-stochastic effects no risk is incurred for exposures below the threshold of observable damage. For stochastic effects a finite probability of damage is assumed down to the lowest levels of exposure. In the following paragraphs these risks are summarized by level of exposure.

Acute whole body doses exceeding 5000 rads cause severe damage to the central nervous system leading to death within 1 or 2 days, or less. Lesser doses exceeding 1000 rad are likely to cause fatal damage to the intestine, and death in 3 to 10 days. In addition, severe depression of the hematopoetic system leads to infection and bleeding, which would cause death within weeks. In either case survival is unlikely.

For acute doses of about 500 rads marrow depression remains severe, but if adequate medical support is provide over a short critical period of 1-2 months, survival is quite possible. If the patient survives no long-lasting damage will be observed, although the risk of cancer, unfertility and damage to the lens of the eye is high.

For acute doses of about 100 rads transient symptoms of shock with nausea, vomiting and headache will be experienced. The probability of survival, however, is very high.

At doses less than 25 rads to the whole body, or 500 rads to a single organ, no lasting effects attributable directly to the radiation exposure will be experienced. Elevated risks of fatal cancer and genetic damage, however, are incurred, similar to those of a worker with a lifetime dose of 25 rads. The excess cancer risk from such an exposure to a typical adult male is about 0.1 to 0.5 percent, with the lower value applicable to a male of about 60 years of age.

The comparable risk to a female is about 50 percent greater, due mainly to the radiosensitivity of the female breast. In comparison the lifetime risk of fatal cancer in the general population is about 20 percent. In addition a small increase in the risk of genetic defects passed on to future generations, by about 0.2 percent, is incurred. The normal incidence of such changes, however, is about 10 percent. At a dose of 25 rads any effects of this exposure would be difficult to detect, apart from small changes in the chromosomes of some cells.

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