POST CHERMOBYL SAFETY REVIEW AT ONTARIO HYDRO

BY

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1.0 INTRODUCTION

It is generally recognized that the Chernobyl Unit 4 accident did not reveal any new phenomena which had not been previously identified in safety analyses (Ref. 1). However, the accident provided a tragic reminder of the potential consequences of reactivity initiated accidents (RIA's) and stimulated nuclear plant operators to review their safety analyses, operating procedures and various operational and management aspects of nuclear safety. Concerning Ontario Hydro, the review of the accident performed by the corporate body responsible for nuclear safety policy and by the Atomic Energy Control Board (the Regulatory Body) led to a number of specific recommendations for further action by various design, analysis and operation groups. These recommendations are very comprehensive in terms of reactor safety issues considered as one can see from Table I.

The general conclusion of the various studies carried out in response to the recommendations, is that the CANDU safety design and the procedures in place to identify and mitigate the consequences of accidents are adequate. Improvements to the reliability of the Pickering NGSA shutdown system and to some aspects of safety management and staff training, although not essential, are possible and would be pursued.

In support of this conclusion, the paper describes some of the studies that were carried out and discusses the findings. The first part of the paper deals with safety design aspects. While the second is concerned with operational aspects.

2.0 SAFETY DESIGN ASPECTS

2.1 Implications of the Chernobyl Accident

A lack of depth of safety analysis and an inadequate shutdown system were clearly important contributing factors to the Chernobyl accident.

Firstly, the design basis for determining shutdown capability did not include all possible power levels and operating modes. Although it was recognized that operation below approximately 20 percent full power was potentially unsafe, conformance with operating procedures was the only means of ensuring adequate safety. Secondly, the Chernobyl designers did not appear to have recognized the potential for high reactivity insertion rates under certain conditions and provided a shutdown system which was clearly too slow. Thirdly, core configurations which could disable the shutdown system were not sufficiently recognized and prevented. Based on these implications, studies were carried out to verify:

a) that the CANDU safety analyses include consideration of the bounding reactivity insertion rate and that shutdown systems, sufficiently effective to terminate RIA's are in place;

b) that there is no credible reactor configuration which could render the shutdown function ineffective; and

c) that the safety of the Pickering NGSA reactors is adequate.

The Pickering NGSA reactors were licensed before there was a requirement for two fully independent shutdown systems and thus it was felt, that they might be more vulnerable than the other Ontario Hydro's CANDU reactors.

2.2 Bounding Reactivity Rate

Three classes of events have the potential to cause a positive reactivity transient in CANDU reactors (Ref. 2). The first class, control failures, involves malfunctions of the reactor regulating system. Reactivity rates up to 2 mk/s are analyzed in the Safety Report although rates in excess of about 1 mk/s are not considered credible. The second class is loss of coolant accidents (LOCA) and, in particular, large break LOCA. Rates of up to 4 mk/s are computed in accident analysis for these accidents. Finally, the third class is loss of coolant flow events for which a mild reactivity transient (< 0.5 mk/s) is possible under certain conditions.

As part of the post-Chernobyl safety review, the control failure sections of the Safety Reports for all the Ontario Hydro stations were completely updated. The analyses covered the full range of possible initial power levels from 10^{-7} to 1.03 full power, and all possible modes in which the reactor can be operated (different configuration and number of heat transport pumps, distorted flux shapes at reduced power, etc.)

In addition, an assessment was carried out to confirm that a large break LOCA from full power conditions indeed bounds the range of uncontrolled positive reactivity insertion rates.

These analyses essentially confirmed the original results and thus the fact that the shutdown systems are sufficiently effective to safely terminate reactivity initiated accidents in all Ontario Hydro reactors. Minor modifications to some of the shutdown system trip setpoints were made to reflect either changes in operating conditions (i.e., increase in reactor inlet temperature) or new experimental information.

2.3 Effect of Initial Core Configuration on Shutdown System Effectiveness

The effectiveness of the shutdown systems can be reduced, as a result of neutronic or process system behaviour, in two fundamental ways. First the initiation signal of a given trip parameter may be delayed by control system action. For instance, reactivity control devices in abnormal position could affect the response of a flux detector employed in the shutdown system. Second, severe flux distortions could result in the reactivity devices of the shutdown system reaching the core region of high neutron importance with some delay, relative to the case of no distortion. For instance, a severe bottom-to-top flux tilt could reduce the effectiveness of the shutoff rods (Shutdown System No. 1). Both aspects of this issue were examined. One assessment addressed potential functional cross-links between either the shutdown systems (for reactors with two shutdown systems) or between shutdown systems and reactor regulating system. In particular the study considered early CANDU designs such as Pickering 'A' in which moderator level control is used as part of the regulating system.

The only potential cross-link identified relates to postulated small break LOCA scenario and, in particular to the limiting case of an in-core LOCA (simultaneous failure of both the pressure tube and the calandria tube) occurring when the moderator contains a high concentration of neutron poison.

For these accidents, as the reactor regulating system attempts to maintain bulk reactor power and compensates for the various reactivity effects introduced by the accident (moderator poison dilution, coolant void, etc.), it also introduces a distortion in the power distribution which could delay the occurrence of process trip relative to the occurrence of fuel sheath dryout. Minor changes in some trip setpoints and the maintenance of a minimum differential between the isotopic purity of the coolant and that of the moderator was sufficient to re-establish adequate trip coverage.

With regard to the effect of flux distortions on the effectiveness of shutdown system, the analyses considered physically possible, not necessarily credible, reactor configurations. Flux tilts considered included those resulting from abnormal position of reactivity devices, Xenon transients as well as operation with low moderator level. These flux distortions were assumed to be present at the initiation of the limiting reactivity accident i.e. large break LOCA. For simplicity in the analysis, the reaching of prompt criticality was used as the criterion to judge acceptable performance. For the conditions of interest, this criterion is bounding with respect to preventing fuel melting or channel rupture.

Typical results for reactors with two shutdown systems are shown in Table II. The Table shows the margin between first negative reactivity effect and prompt criticality. A positive margin indicates that the shutdown system remains effective. Results are presented for the Pickering 'B' reactors (550 MW(e)) and the Darlington reactors (881 MW(e)). The Table shows that this is a considerable margin.

For the Pickering NGSA which have one shutdown system and moderator level control it was found that the shutoff rod system remains effective for all RIA's as long as the flux distortion is no worse (in terms of bottom-to-top tilt) then that induced by operating at a moderator level of 68 percent (seven rows of fuel channels uncovered).

Although the combination of a large break with ope: ation in a severely distorted flux shapes is considered incredible (see Section 2.4 below), an automatic reactor trip to prevent this mode of operation is being installed. The conceptual design of the trip is discussed later in this paper.

2.4 Safety of the Pickering 'A' Reactors

As indicated above these reactors have a single shutdown system which consists of shut off rods supplemented by moderator dump. The studies carried out for these reactors are therefore of particular interest to operators of older PHWR plants.

As part of the post-Chernobyl safety review, a number of separate studies were carried out with the general purpose to quantify the level of redundancy of the shutdown system, its reliability and the consequences of failure to shutdown following a large break loss of coolant.

To quantify the redundancy two studies were carried out. The first one determined the minimum number of shutoff rods that need to be available in order to safely terminate a large break LOCA. The results are shown in Fig. 1. The Figure illustrates that even for the limiting LOCA (guillotine failure of the Reactor Inlet Header) six "most effective" or 11 "average" shutoff rods, out of 21 provided, can be unavailable without compromising the effectiveness of the shutdown to the extent that fuel channel integrity is not maintained. The second study quantified the capability of the moderator dump acting alone to terminate reactivity transients. It concluded that the moderator dump is capable of terminating all RIA's with the exception of a large break LOCA.

A probabilistic study was carried out to assess the frequency of a loss of shutdown event. The results are summarized in Table III. The Table shows that the total frequency of events involving failure to shutdown is 6.2×10^{-6} reactor - yr^{-1}. In particular, the frequency of shutdown failure combined with a large break LOCA is estimated to be in the meaningless range of 10^{-9}/reactor - yr^{-1}.

Events of this frequency are normally considered incredible and their consequences are not analyzed in detail. Nevertheless, an analysis was performed of an event involving the limiting large break LOCA coincident with shutdown failure (Ref. 3). The conclusions are that despite severe damage to the reactor core, the structural integrity of the containment envelope is maintained and the off-site consequences are not expected to be significantly worse then those calculated for low frequency design basis event such as LOCA with coincident failure of emergency coolant injection.

3.0 DESIGN MODIFICATIONS

The analyses described led to the conclusions that safety related modifications for CANDU reactors with two independent shutdown systems are not warranted. In particular, disabling of the shutdown function as a result of physically conceivable flux distortions or tilts is virtually impossible. For the Pickering 'A' reactors however, flux tilts large enough to disable the shutdown function are conceivable, although extremely unlikely as discussed in Sect. 2.4. Both automatic features and administrative controls are in place to prevent operation with large flux tilts. However, operation at reduced power with severe flux tilt cannot be completely precluded by automatic system action alone. Therefore it was decided to install an additional shutdown system trip to ensure automatic reactor shutdown for unacceptable flux tilts. Furthermore, as a result of various discussions with the Regulatory Body a decision was taken to further improve the reliability of the shutdown system by adding a complete set of trip parameters to the existing ones. The conceptual aspects of these modifications are briefly discussed in the following.

3.1 Flux Tilt Trip

A schematic representation of the flux tilt trip is shown in Fig. 2. Three pairs of in-core detectors will be installed in three existing calandria penetrations. Each pair of detectors will be located at approximately 230 cm below and above the centre of the core. For each pair of detectors the ratio of the signal from the bottom and top detectors is compared to a setpoint. If the ratio exceeds the setpoint, a logic channel is trip, trip of two channels leads to a reactor trip.

The setpoint is selected to provide a reactor trip whenever the flux tilt in the core approaches that induced by a moderator level of 68 percent. Suitable allowances are made for calculation and calibration uncertainties. The trip is conditioned in automatically and may be conditioned out manually once the reactor power is at or below 12 percent. This allows operation at low moderator level at reactor power low enough that disabling of the shutdown system by a severe flux tilt is not possible. The trip remains effective for all operating modes at high power such as operation with adjuster rods withdrawn for shim and poison override, etc.

The flux tilt trip also provides an effective trip signal for moderator pipe failures. Following such failures the Reactor Regulating System (RRS) will attempt to maintain the reactor power at the initial level. However, the combined action of the RRS and of the decreasing moderator level may lead to overpower in some low elevation fuel channels. The flux tilt trip is effective in preventing fuel sheath dryout for this scenario.

3.2 Additional Set of Trips

The Pickering 'A' shutdown system comprises one set of triplicated detection, relay and instrumentation system for each trip parameter. Any trip parameter exceeding a pre-calculated value causes the rapid insertion of 21 shutoff rods. These rods provide sufficient negative reactivity rate and depth to safely terminate any credible RIA's.

Should the rate of power decrease be slower than a predetermined reference rundown, a separate shutdown mechanism is activated whereby the moderator is quickly removed from the calandria vessel.

The modified system is illustrated in Fig. 3. The detection instrumentation, alarm unit and logic for each trip parameter will be duplicated and separated into two logic chains. Any trip parameter being exceeded in any of the two logic chains will initiate insertions of all the shutoff rods. The moderator removal logic remains unchanged. These modifications will increase redundancy and improve tolerance to common mode failure as previously discussed.

4.0 OPERATIONAL PROGRAM ASPECTS

4.1 Review of NGD Fire Fighting Capability

As a result of the explosion and subsequent fire during the Chernobyl accident, lethal doses of radiation were received by fire fighting personnel.

The magnitude of the accident resulted in Ontario Hydro senior managements' initiation of an assessment of fire prevention and fire fighting protection programs in place at Ontario Hydro's nuclear generating stations. The objective of the assessment was to determine the adequacy of controls to maintain the fire risks to in-station personnel and back-up fire fighting forces at an acceptable level by recommending improvements to ensure adequacy of the programs.

Both fires with the potential to be accompanied by radiation hazards and those of a purely conventional nature were assessed. The assessment was carried out by a task group consisting of Ontario Hydro and Atomic Energy of Canada Ltd. staff. Recommendations of the task group were reviewed and endorsed by the Ontario Fire Marshall's Office.

Existing fire safety programs were analyzed for compliance with applicable legislation, fire codes and industry standards. The task group also use professional judgement in recommending further improvements and program good practices. The findings of the task group concluded that the design basis for fire fighting systems in the stations is such that automatic extinguishing systems are provided where high fire loads and/or risks to fire fighters are perceived. In other areas, manual systems are provided. In addition, the spatial separation of redundant safety systems and the fail safe mode of shutdown systems, largely reduce the probability of fires jeopardizing safety systems.

Such inherent features contribute to reducing the probability of a major fire occurring within the stations. Furthermore, after detailed analyses of station systems and operations with fire hazard potential, the fire prevention and protection systems provided, and the management controls in place, it may be concluded that the probability of in station fire fighters being exposed to radiation doses significantly greater than legal limits is low.

Notwithstanding these facts, the task group did find that the Ontario Hydro stations need to improve aspects of both the managed system and hardware systems of their fire safety programs.

While many identified program deficiencies have been corrected to date, further efforts to determine hardware requirements commensurate with risks to both reactor and worker safety, and material losses, are currently underway.

4.2 An Assessment of Ontario Hydro's Operating Nuclear Reactor Safety Systems - Accessibility of Protective Circuits

To provide assurance that the controls in-place to limit accessibility to the reactor safety systems are adequate, quantitative reviews were initiated to address reactor safety issues which had been violated at Chernobyl. The following two issues were reviewed:

a) Bypassing of reactor trips associated with steam separator pressure and water level which enabled the reactor to be operated in an unstable condition, and

b) Control rods had been withdrawn well beyond safety limits specified by procedures to compensate for xenon buildup, and the emergency core cooling systems had been deactivated for over 9 hours while the plant was operating.

In order to ensure that safety systems could not be bypassed or deactivated as described in a) and b) above, the assessment focused on the ways in which CANDU rector safety system circuitry could be disabled and the effectiveness of controls in place to prevent either deliberate or inadvertent disabling.

The study was undertaken in two phases. One phase reviewed the barriers in place to prevent deliberate disabling of safety systems, while the other phase reviewed the barriers in place to prevent inadvertent disabling of safety systems. Specifically, the systems considered in the review were the shutdown systems, the emergency coolant injection system, the containment system and the stepback and setback systems. For the purpose of the study all safety system impairments were considered disablements. Malicious disabling was not considered in this study.

The assessment addressed system disablement of the following characteristics:

A safety system being deliberately disabled as a result of circuit logic being blocked or placed in a state in which it could not operate if required, or by individual parameter setpoints being adjusted in the unsafe direction such that an individual parameter either would not trip at all, or would not trip at the required setpoint.

The assessment methodology involved:
• a detailed review of the physical and administrative controls in place to prevent the systems from being deliberately disabled, as well as
• a detailed review of the physical and administrative controls in place to prevent the systems from being inadvertently disabled.

The overall finding of the assessment was that there is no evidence to suggest that there is significant potential of safety systems from being disabled, the number of physical barriers is limited. Another significant finding of this particular study was that at most stations, no test had been specified following the calibration of conditioning signals.

The comparison of barriers in place with the composite sets identified that in some cases minor deficiencies or inconsistencies exist in administrative controls. Recommendations have been made wherever a deficiency or inconsistency was found in order to further strengthen existing barriers. A general recommendation requiring the development of a set of standards for the in-place administrative barriers to prevent disablement was also made. Work on the development of such standards is presently in progress.

4.3 Operational Reactor Safety Management Model

The management of operational reactor safety was reviewed to determine the effectiveness of the system and to further clarify and communicate the roles of groups and individuals associated with the function. The model was reviewed to ensure that all aspects of the reactor safety program were adequately addressed.

The review systematically examined the rigor of in-place models by examining managed system elements required to ensure adequate reactor safety. The existence and effectiveness of the following key managed program elements were reviewed:

- a clearly defined inviolable operating envelope
- clearly defined responsibilities and limits of authority
- qualification of staff performing work activities
- use of approved equipment and materials
- documentation of work activities
- surveillance program execution
- emergency preparedness procedures

The requirements for each of these elements as they apply to an adequate model and the encompassing activities such as training, supervision, work planning and audits were documented in detail.

Responsibilities and limits of individuals, groups and/or agencies associated with the safe management of nuclear reactor operation were also documented. With specific reference to the Chernobyl accident, the training of panel operators and shift-supervisors-in-training has been augmented with additional modules which specifically address appropriate responses to prompt criticality events, and the reactivity effects of coolant void.

With respect to the functioning of the model as it impacts on nuclear safety, an examination of station policies and procedures was performed at all stations. This study verified that at all stations work plans and revised test procedures are reviewed by at least one senior individual from both the Technical and Production sections, all of whom have been sufficient knowledge of system design, interaction and safe operating practices to prevent a procedure which may compromise reactor safety, from being implemented.

4.4 Off-site Preparedness Review

In Canada, responsibility for off-site preparedness rests with the Provincial Government. Following the Chernobyl accident a Working Group was constituted under the authority of the Ontario Ministry of the Solicitor General. The Working Group was tasked with developing reference accidents which would serve as a basis for assessing requirements for off-site emergency response capability. As such the Working Group was required to make recommendations on

a) an upper limit for detailed emergency planning and preparedness in Ontario given in terms of the radiological situation off-site, and

b) the extent to which mitigation measures needed to be adopted as a result of the upper limit recommended by the working group in addition to those prescribed in or under the then current Provincial Nuclear Emergency Plan.

The scope of the recommendations was expanded to include both an upper limit for planning, a scientific assessment of the risks of various types of accidents, and also risks which cannot be scientifically assessed, such as those due to hostile action.

The Working Group was made up of 6 persons well versed in emergency planning nuclear power, or health physics. One manager level Ontario Hydro employee served as a member of the Working Group. Based on all of the above recommendations the Working Group recommended that the size of the Primary Zones for each site should be 13 km.

Further recommendations included that the Province consider taking appropriate measures in the following areas:

- the availability and distribution of potassium iodide pills
- the need for early warning systems for the public
- the need for adequate medical facilities to deal with possible acute radiation exposure
- the advisability of restricting new housing construction near nuclear facilities
- review the present Protective Action Levels (PALs) in the light of world norms.

The Working Group emphasized that their terms of reference specifically excluded consideration of social, political and economic factors and the recommendations were based purely on technical considerations. At this time, revisions to the existing Provincial Nuclear Emergency Plan are being formulated by the Solicitor General and are expected to be approved by the Provincial Cabinet before mid-year.

4.5 Source Term Estimation

In conjunction with the development of reference accidents by the Provincial Working Group, Ontario Hydro continues to develop computer programs for real-time use following accidental airborne releases of radioactive materials, both to predict public doses and to better support decisions regarding public protection strategies.

The computer programs in use at Ontario Hydro for these eventualities are a series of Emergency Response Projection (ERP) codes. ERP codes have been available for the Pickering and Bruce sites for some time. With the coming on-line of the Darlington station, an improved version of these codes was developed. The Darlington Emergency Response Projection code (DERP) marked the beginning of a new generation in the series of ERP codes, making it the standard to which the older codes will be converted to in the near future.

DERP uses a modular structure to improve flexibility of operation, makes fuller use of post-accident measurements and includes a broad scope of predictive capabilities. In addition, it accommodates both expert and non-expert users. The enhancements in DERP are primarily aimed at providing the expert user with greater flexibility to exercise judgement and modify built-in assumptions and data. The non-expert user also benefits from improved displays and a user friendly environment.

From a source term estimation point of view, DERP performs two types of functions: a) it predicts the source term based on fission product, tritium and particulates models adjusted for current plant conditions, and b) it allows incorporating measurements of specific in-plant parameters to iteratively adjust the predicted source term to make it more closely correspond with observed parameters.

The process of adjusting and predicting the radiological source term by incorporating the latest available in-plant information is referred to in ERP as Source Term Adjustment. Two types of in-plant information are used for Source Term Adjustment purposes: Post Accident Radiation Monitoring System (PARMS) measurements, and gamma survey measurements.

PARMS measurements are acquired from samples obtained at special take-off points in the Emergency Filtered Air Discharge system. These samples provide a direct measurement of the radioisotopes in the exhaust effluent. Gamma survey measurements are obtained by measuring gamma dose rates at pre-selected locations outside containment with known geometry and shielding. In this case, the code assumes that the measured fields result from fission products in the containment atmosphere of known composition, released during the accident.

From both types of measurements the code derives scaling factors to adjust the predicted source term. In the case of PARMS, scaling factors are applied to individual isotopes. For gamma survey measurements, generalized scaling factors are used.

Once th‿ source term is estimated, other mr lules in the code evaluate dilution factors, potential public doses, times to reach various Protective Action Levels and worker doses incurred in carrying out emergency response operations. This information allows to better support decisions regarding public protection strategies.

4.6 <u>In Station Worker Protection</u>

In addition to information gathered from the initial review of the Chernobyl accident, recommendations by Canada's federal nuclear regulator, the Atomic Energy Control Board (AECB), also provided an impetus for Ontario Hydro to review in-station worker protection during emergencies.

In-plant worker protection during a radiological emergency is modelled by the key response elements and programs, many of which have been and are being developed on the basis of recommendations of other post Chernobyl reviews.

Specifically, the actions outlined are aimed at response programs to better define objectives, enhance current performance and prepare some key contingencies for potential severe radiological accidents. Three major areas for consideration have been identified:

1. There are a number of issues which have a direct effect on personnel safety policy and the development of implementing procedures. For example:

- There is a need to have a more explicit definition of the scope of some of the emergency program elements. Without this, the basis for providing appropriate expectations and training is not defendable.

- Some response programs lack the flexibility to ensure that they can be readily expanded in times of need. Formal interfaces are needed at both local and corporate levels to ensure that response capability can be supported with additional resources.

- Programs are slow in implementing new technology and to utilize operating experience effectively to improve performance

The drill and exercise program does not adequately test performance of all the key worker protection response programs.

2. The study identified two hardware related issues:

Inventory control of supplies and personnel protective equipment and the availability of radiological survey meters in most response programs, was found to be less than adequate.

The design of some systems and radiological measurement instrumentation does not utilize cost-effective advancements in technology.

3. The manner in which the priorities of the emergency are established, the timeliness of actions, the contamination control philosophy, and the exposure control and safety philosophy are not always conducive to effective overall performance during a nuclear emergency. Better ways of integrating the procedures, hardware and the emergency environment for timely effective response need to be developed.

In order to address the collective issues impacting on in-station worker protection, nine key initiatives are recommended to upgrade current capability and performance. These are:

- systematic redevelopment of emergency procedures on an as required basis to adequately reflect program expectations, scope, expansibility and interface with external resources
- implementation of upgraded, formal training and qualification programs for emergency response functions
- establishment of comprehensive drill and exercise programs which will routinely test and maintain a high level of emergency response capability
- evaluation of the development of a personal electronics safety device, capable of integrating several key emergency response functions such as accident dosimetry, personnel alerting/communications etc., with routine requirements
- documentation of the current personnel radiological accident dosimetry program used by Ontario Hydro
- evaluation and recommendation of action plans for computerized personnel accounting
- redefining and implementation of an effective emergency contamination control program, and
- development and implementation of a system for rapid identification and access to resources and technology to support emergencies at Ontario Hydro nuclear facilities

All of these initiatives are presently underway in Ontario Hydro.

5. Conclusion

The potential for positive reactivity excursions exists in all types of commercial reactors. The most effective way of precluding this type of accident is to maintain the defence in depth approach. This was recognized very early in the development of the CANDU design and led to a combination of inherent and engineered safety features such that the public risk associated with reactivity accidents is negligible. The review of the implications of the Chernobyl accident on the safety design of the Ontario Hydro CANDU reactors confirmed this conclusion. For the Pickering 'A' reactors, the review concluded that an automatic means of precluding operation with highly distorted flux shapes is desirable. A flux tilt trip has been designed for this purpose as well as to provide additional coverage for moderator pipe failures. The review of the implications of the Chernobyl accident on the operational aspects did not uncover any major deficiency. A number of improvements were however identified and are being implemented. These cover aspects such as fire fighting capability, control to prevent disabling of safety systems, emergency planning response and in-station worker protection.

References

1. INSAG-1, IAEA, Vienna 1986

2. Morison, W.G., et al., "Dealing with Reactivity Initiated Accidents in Commercial Power Reactors", ENS/ANS International Conference on Thermal Reactor Safety, Avignon, France, October 1988

3. Luxat, J.C., "Insights to the Phenomenology and Energetics of Reactivity Initiated Accidents", ENS/ANS International Conference on Thermal Reactor Safety, Avignon, France, October 1988

Table I

Issues Considered in the Post-Chernobyl Review

- shutdown systems capability for all situations
- safety of the Pickering 'A' reactors (single shutdown system)
- Containment capability for Severe Reactivity Initiated Accidents
- Accessibility of reactor protective circuits
- Safety management review
- Training and informing of operators
- Off-site preparedness review
- Firefighting capability
- In-station worker protection
- Source term estimation for emergency response
- Review of operating procedures

Table II

Margin Between First Negative Reactivity
Effect and Prompt Criticality for Severe Flux Tilts

Bottom Over Top Flux Tilt

Reactor	Shutdown System No. 1 (Expected Performance)[1]	Shutdown System No. 1 (Licensing Assumption)[2]	Shutdown System No. 2 (Licensing Assumption)
Pickering 'B'	0.56s	0.23s	0.85s
Darlington	(3)	2.7s	>>0.6s

Side to Side Flux Tilt

	Shutdown System No. 1 (Licensing Assumption)	Shutdown System No. 2 (Licensing Assumption)
Pickering 'B'	0.65s	1.00
Darlington	>0.41s	3.2s

Notes
(1) Calculation Performed using measured (expected) shutoff rod drop curve
(2) Calculation performed using shutoff rod drop curve assumed in licensing analysis
(3) Not calculated since margin with licensing assumption is large

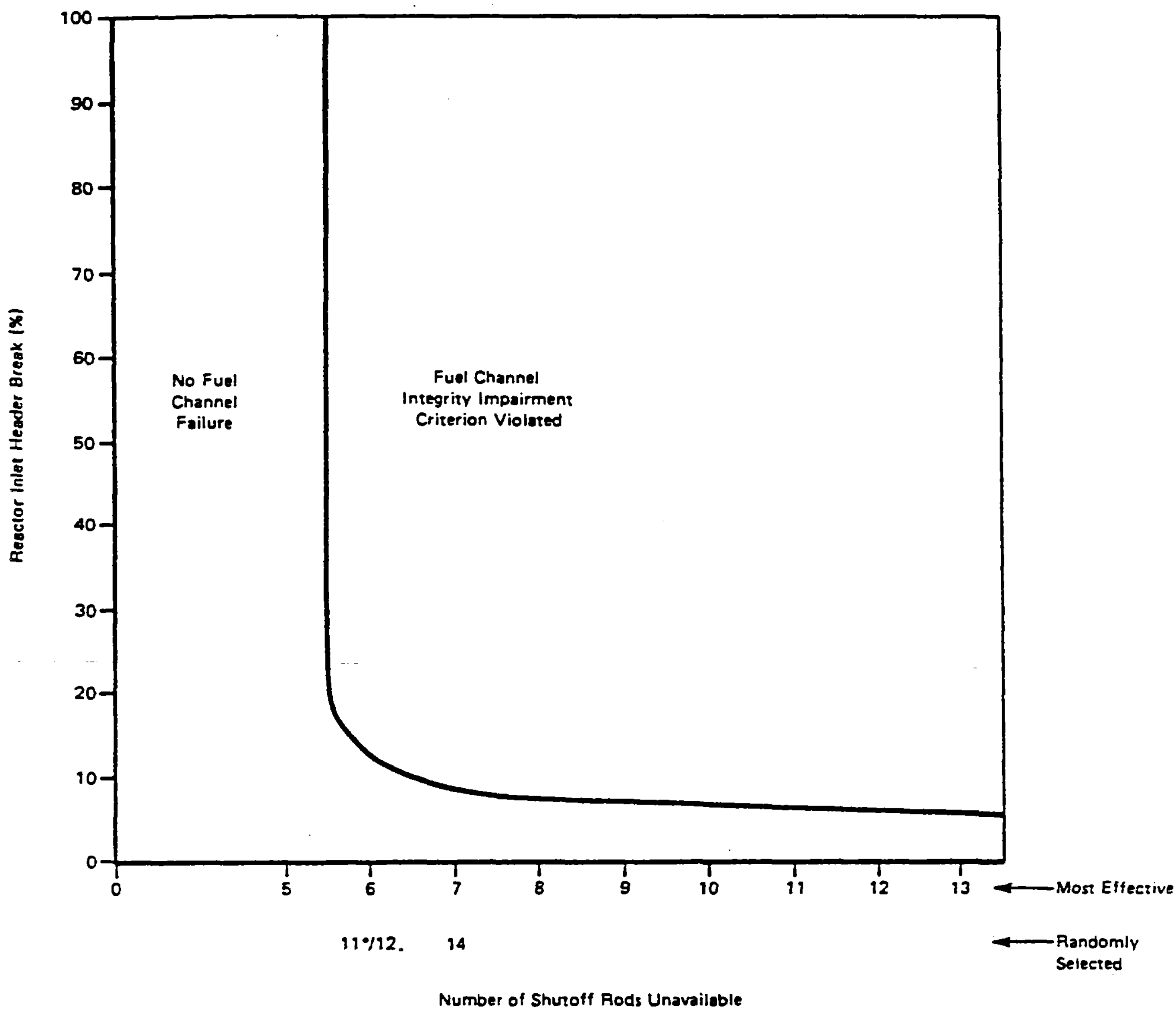
Table III

FREQUENCY OF FAILURE TO SHUTDOWN (WITH PROB. CUT-OFF)

Initiating Event/ Frequency (Reactor-yr^{-1})		Probability of Shutdown Failure	Event Sequence Frequency (Reactor-yr^{-1})
1.	Fast Loss of Regulation (2×10^{-2})	10^{-5}	2×10^{-7}
2.	Medium Loss of Regulation (3×10^{-2})	10^{-5}	3×10^{-7}
3.	Slow Loss of Regulation (3×10^{-2})	1.1×10^{-5}	3.3×10^{-7}
4.	Large LOCA (3×10^{-4})	1.6×10^{-5}	4.8×10^{-9}
5.	Small LOCA (2×10^{-2})	10^{-5}	2×10^{-7}
6.	Loss of Cl. IV (2×10^{-2})	10^{-5}	2×10^{-7}
7.	Sym. Loss of Feedwater (1.1×10^{-1})	10^{-5}	1.1×10^{-6}
8.	Asym. Loss of Feedwater (6×10^{-2})	10^{-5}	6×10^{-7}
9.	Asym. Loss of Feedwater (6×10^{-3})	1.2×10^{-4}	7.2×10^{-7}
10.	Pressure Tube Failure (2×10^{-2})	10^{-5}	2×10^{-7}

Table III (cont'd.)

Initiating Event Frequency (Reactor-yr^{-1})		Probability of Shutdown Failure	Event Sequence Frequency (Reactor-yr^{-1})
11.	Steam Gen. Tube Break (1×10^{-3})	10^{-5}	1×10^{-8}
12.	High Heat Transport System Pressure (9×10^{-2})	10^{-5}	9×10^{-7}
13.	Low Heat Transport System Pressure (1.3×10^{-1})	10^{-5}	1.3×10^{-6}
14.	Loss of Mod. Inventory (1.2×10^{-2})	10^{-5}	1.2×10^{-7}
15.	Steam Line Break (4×10^{-3})	10^{-5}	4×10^{-8}
		TOTAL	6.2×10^{-6}



* For 100 Percent RIH Break

FIGURE 1
Fuel Channel Integrity Impairment Envelope

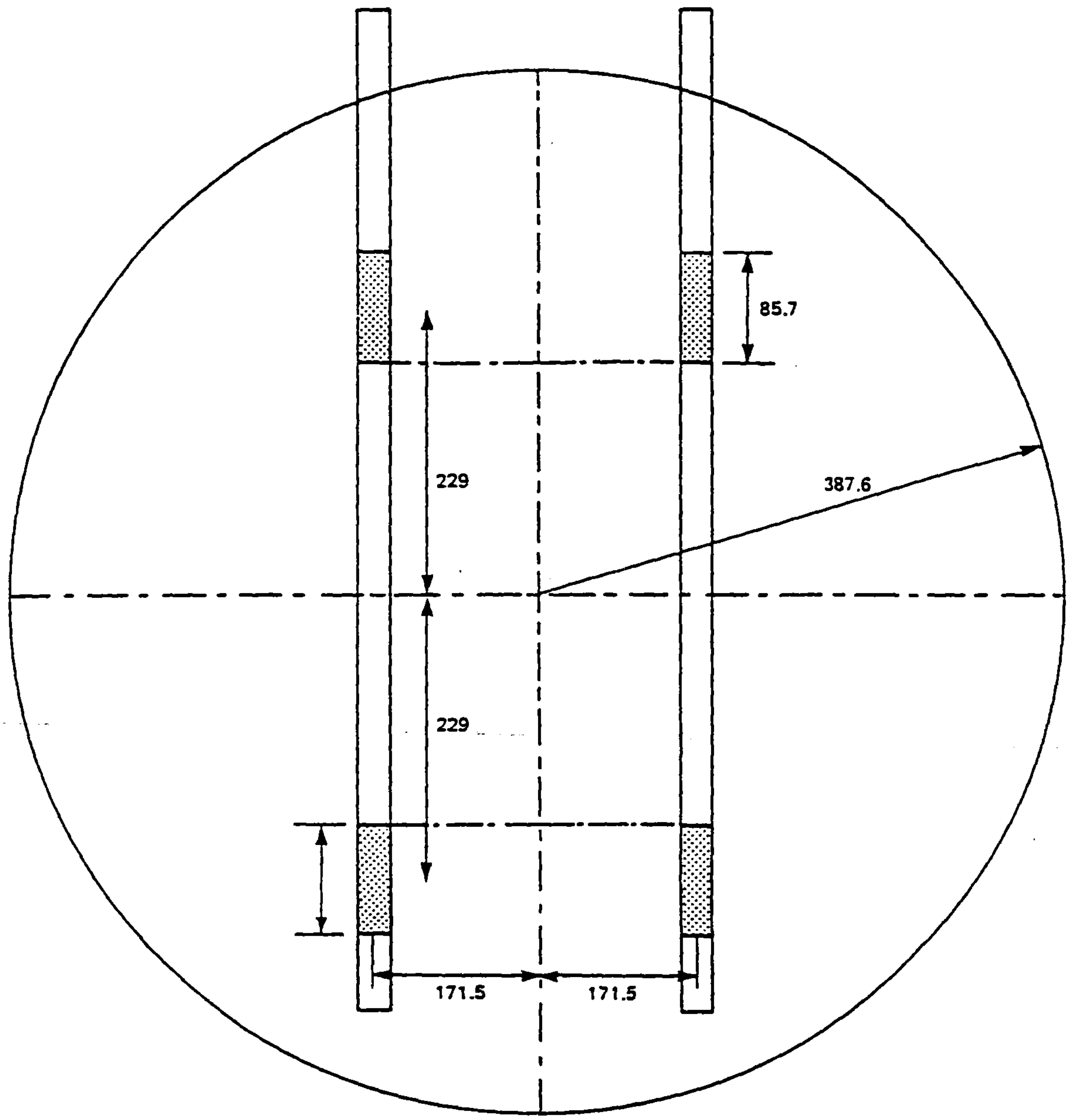


Figure 2

Detector Locations for Flux Tilt Trip and Low Moderator Level Overpower Trip
(Units cm)

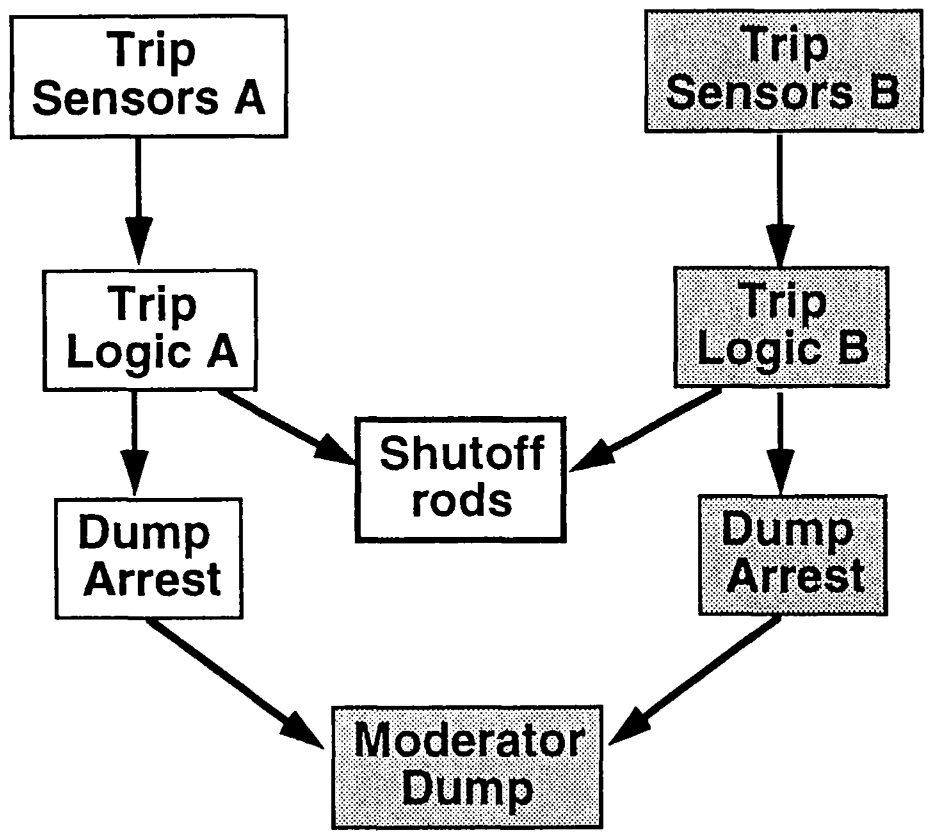


FIGURE 3 : SCHEMATIC REPRESENTATION OF THE PICKERING 'A' SHUTDOWN SYSTEM