INDIAN POINT UNIT NO. 2

EVALUATION OF THE ADEQUACY OF EXISTING NEUTRON FLUX INSTRUMENTATION FOR NUREG-0737, SUPPLEMENT 1

Prepared For: Consolidated Edison Company of New York, Inc.

Prepared By:

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TABLE OF CONTENTS

		Page
1.0	INTRODUCTION	1-1
2.0	HISTORICAL PERSPECTIVE	2-1
3.0	SAFETY ASSESSMENT	3-1
3.1	Design Basis Accident Analysis Evaluation	3-1
3.1.1	Loss of Coolant Accident (LOCA)	3-2
3.1.2	Main Steam Line Break (MSLB)	3-6
3.1.3	All Other Design Basis Accidents	3-8
3.2	Accidents Beyond Design Basis	3-9
3.3	Boration Requirements	3-11
3.4	Assessment Results	3-12
Table 3-1:	Functional Failures	3-13
Table 3-2:	Beyond Design Basis Event Scenarios	3-15
Figure 3-1:	Typical Graph of Minimum Shutdown Boron Concentration Versus Cycle Burnup	3-17
4.0	COST-BENEFIT ANALYSIS	4-1
4.1	Approach and Method	4-1
4.2	Results Table	4-2
4.3	Results Table Discussion	4-3
4.4	Conclusion	4-6

TABLE OF CONTENTS
(continued)

		Page
Table 4-1:	Functional Response Items Considered	4-7
Table 4-2:	Cost Elements Considered	4-8
Figure 4-1:	Neutron Flux Monitoring System with Appendix R Interface	4-9
5.0	BASIS FOR EOP TECHNICAL GUIDELINES	5-1
5.1	Subcriticality Criteria for EOP Technical Guidelines	5-1
5.2	Assessment of EOP Technical Guidelines	5-13
6.0	POST ACCIDENT SAMPLING SYSTEM REVIEW	6-1
7.0	EMERGENCY BORATION	7-1
7.1	Control Room Walkthrough	7-1
7.2	Availability and Reliability Assessment	7-1
8.0	CONCLUSIONS	8-1
9.0	REFERENCES	9-1

APPENDICES

APPENDIX A - NUREG-0737, Supplement 1, References to Guidance Documents

APPENDIX B - NUREG-0737, Supplement 1, References to Integration of Initiatives

APPENDIX C - Excerpts from Official Transcript of 2/22/83 Supplement 1 Regional Workshop

APPENDIX D - Reprint of Appendix RECRIT from NSAC-1 Supplement

1.0 INTRODUCTION

As a result of the TMI-2 accident many new requirements were generated to improve accident detection and mitigation capabilities of nuclear power plants. During the accident itself there was much confusion in the Control Room regarding the operator's information needs, especially as plant conditions were changing. Many industry and NRC sponsored studies were done after the accident to better define the role of the operator and the man-machine interface. The possibility of using source range neutron flux instrumentation to gauge the level of voiding of the core due to bulk boiling or to gauge the degree to which the core is uncovered arose from the TMI-2 incident. Correlation of various plant conditions with the response of the source range instrument led to the tentative conclusion that actual core water level during uncovery could be related to changes in the source range count level but the phenomenon was not well understood. The events that involve core uncovery are typically loss of coolant accidents (LOCA) and the neutron flux monitoring instrumentation would have had to provide meaningful information during such an event. This would have required an extensive performance testing program under various hydraulic and environmental conditions. Additionally, when the TMI action plan was at its peak of implementation, an event involving a delayed reactor trip occurred at the Salem plant. This fueled the long-standing debate over the Anticipated Transients Without Scram (ATWS) issue and led to additional programs, some of which impacted the man-machine interface initiatives, particularly with regard to implementing ATWS response procedures.

At that time, the Westinghouse Owners Group (WOG) had already formulated ATWS recovery procedures within the Emergency Response Guidelines in the form of emergency instruction ECA-1, entitled "Anticipated Transients Without Scram." This was later integrated into the WOG guideline set as procedure FR-S.1 and ECA-1 was then eliminated as an ATWS recovery procedure. The ATWS events analyzed by WOG did not involve adverse environmental conditions inside containment and the existing neutron flux instrumentation was thus considered suitably qualified for use with procedure FR-S.1. The ATWS procedures fully intended to use existing neutron flux instrumentation as an indicator that an ATWS is occurring and to provide information to assist an operator in its mitigation. Since the neutron flux instrumentation was capable of fulfilling this intended design and safety function, there was never any need to upgrade or change it out for the purpose of procedure implementation. Any need to upgrade the neutron flux instrumentation as a result of using it to measure core water level during a LOCA also became unnecessary since other suitable water level measuring systems were installed in Westinghouse NSSS plants. Hence, the neutron flux instrumentation as originally designed was intended to fulfill its function in the current design basis accident mitigation strategies.

The Regulatory Guide 1.97 evaluation program followed by Con Edison for the selection and implementation of post-accident monitoring instrumentation is plant specific owing to differences in plant design, operating philosophy, integration of Supplement 1 initiatives, and numerous other factors. An engineering evaluation of the instrumentation for each plant included a review of Regulatory Guide 1.97, Revision 2, an evaluation of the plant specific accident monitoring needs, procedures and a review of existing instrumentation. This resulted in the development and documentation of plant specific justification of existing equipment and modifications or addition of equipment (with its justification) where necessary. This effort determined that it is <u>not</u> necessary to upgrade neutron flux instrumentation because an integrated assessment of NUREG-0737, Supplement 1 shows that ***there is no accident that yields an adverse containment environment that also requires neutron flux to function as a reactor protection circuit.***

Indian Point 2 (IP-2) has since been requested by the NRC to make a commitment to upgrade neutron flux instrumentation to Category 1 from its existing status of Category 3. Con Edison has conducted an in-depth technical and safety review to further document the basis for concluding such an upgrade is not warranted and that the requested exception to Regulatory Guide 1.97, Revision 2 be granted.

This report presents a regulatory and technical analysis of this issue as well as development of an emergency operating procedure solution that utilizes other direct reading instrumentation, which was installed and accepted per NUREG-0737, Supplement 1, to conservatively determine the subcritical status of the core for both design basis and beyond design basis events that cause adverse containment conditions. Additionally, since an adverse containment condition is caused by a primary or secondary system pipe rupture inside containment, the same events that cause predicted core voiding and/or core uncovery, this report presents an evaluation that shows neutron flux information can be misunderstood, precisely the man-machine interface conditions that Supplement 1 is designed to address. This evaluation includes conditions not necessarily anticipated following standard event analysis defined paths. The review shows that even for events beyond the design basis, core exit and RCS temperature monitoring are more meaningful in determining the status of the core and that flux monitoring instrumentation will behave erroneously and can be misleading. Therefore, in keeping with NUREG-0737 Supplement 1 requirements, this report concludes that ***no safety benefit can be found by upgrading neutron flux to Category 1.*** When this conclusion is factored into the cost-benefit analysis, a cost of over $3 million for the upgrade in the IP-2 plant is not justifiable.

2.0 HISTORICAL PERSPECTIVE

The development of SECY-82-111, the predecessor to NUREG-0737, Supplement 1, made it clear that the large body of guidance documents that were to be implemented to improve the man-machine interface would be evaluated as a whole and utilities would be given implementation flexibility. When NUREG-0737, Supplement 1 was issued it stated; "It is not intended that these guidance documents (NUREG reports and Regulatory Guides) be implemented as written; rather, they should be regarded as useful sources of guidance for licensees and NRC staff regarding acceptable means for meeting the fundamental requirements contained in this document." Therefore, using a direct reading, unambiguous, alternative method that does not rely on neutron flux instrumentation to confirm that the core is subcritical for an event that creates an adverse containment environment, should be acceptable under the requirements of Supplement 1. This is especially important because ***events leading to adverse containment environments typically involve core voiding and core uncovery, those same events that have been analyzed to show that neutron flux readings can be misunderstood.*** Having a method for determining the subcritical status of the core that is applicable to adverse containment environment conditions is consistent with the fundamental requirements of NUREG-0737, Supplement 1 and the Commission's policy with regard to upgrading the man-machine interface. This policy was further clarified at NRC sponsored workshops held in February, 1983 with respect to post accident monitoring guidance contained in Regulatory Guide 1.97, Revision 2 wherein the NRC stated that deviations should be explicitly shown and supporting justification or alternatives should be presented.

On December 17, 1982 the Nuclear Regulatory Commission issued Supplement 1 to NUREG-0737 (Generic Letter No. 82-33). Supplement 1 set forth the requirements for emergency response capability in nuclear power plants basically by improving the man-machine interface. The letter is a distillation of basic requirements from the broad range of guidance documents that had been issued by the NRC at that time. In several places Supplement 1 highlighted two important facets to achieve implementation. First, the generic letter and enclosures stressed that the guidance documents (principally NUREG reports and Regulatory Guides) are not to be used as requirements by NRC reviewers and Licensees. Excerpts from Supplement 1 and Generic

Letter No. 82-33 regarding the use of guidance documents are presented in Appendix A. It was noted that Regulatory Guides such as Regulatory Guide 1.97, Revision 2 are to be treated as guidance. Second, the generic letter stressed that a phased integrated program be established to assess the following initiatives:

1. Safety Parameters Display System,
2. Detailed Control Room Design Review,
3. Regulatory Guide 1.97 (Revision 2) - Application to Emergency Response Facilities,
4. Upgrade of Emergency Operating Procedures,
5. Emergency Response Facilities, and
6. Meteorological Data.

Excerpts from Supplement 1 and the Generic Letter regarding the NRC's request to address these initiatives as an integrated program are presented in Appendix B. It was noted that decisions on upgrading plant equipment should be a result of an integrated assessment.

Supplement 1 went on to say that licensee questions regarding Commission policy on these issues would receive responses at regional workshops conducted by senior staff members. Excerpts from the transcript of the February 22, 1983 Regional Workshop are presented in Appendix C. It was again stressed that Regulatory Guides are to be considered guidance. Also, it was noted that use of emergency operating procedures that include how certain instruments are to be used would be an acceptable basis rather than selecting instruments based on using Regulatory Guide 1.97, Revision 2 as a punch list.

The typical methodology for the Regulatory Guide 1.97, Revision 2 review used at the IP-2 plant included:

(1) A survey of the control room instrumentation and the SPDS parameters;

(2) A review of the regulatory guide to develop a list itemizing types and categories of variables that are recommended;

(3) A review of control room instrumentation and emergency operating procedures usage;

(4) A review of the plant instrumentation documentation to evaluate its capabilities and degree to which it meets each Regulatory Guide recommendation;

(5) Preparation of design change packages for modifications; and

(6) Preparation and submittal of a report summarizing the integrated assessment performed, committing to certain instrumentation upgrades or additions, and presenting technical justification for existing plant instrumentation found acceptable.

Regulatory Guide 1.97, Revision 2 was used as a generic source of guidance for this evaluation. The review of the variables classified them into the five types A, B, C, D and E as defined by the regulatory guide. Type A variables were derived from the Emergency Operating Procedures.

Guidelines associated with control room layout and design and with human factors engineering considerations were coordinated with the Regulatory Guide 1.97, Revision 2 review and the other Supplement 1 initiatives.

Previously completed and on-going control room studies and modifications were utilized to assist in the Regulatory Guide 1.97, Revision 2 review. Any recommendations for additions to, deletions from, or changes to the control room instrumentation were designed with the principles of human factors engineering and coordinated with the control room design review programs.

Emergency response capabilities were integrated with the Regulatory Guide 1.97, Revision 2 review and coordinated with the other initiatives of Supplement 1 to NUREG 0737 in order to optimize the interface requirements.

The survey of instrumentation included analysis of the extent to which it has been qualified for post-accident monitoring. The analysis associated with the Regulatory Guide 1.97 efforts were coordinated with ongoing programs to ensure that consistent equipment qualification criteria were applied.

As suggested by Supplement 1 to NUREG 0737, a summary table was prepared which included, for each Type A, B, C, D, and E variable, the following:

- Instrument range;
- Environmental Qualification;
- Seismic Qualification;
- Redundancy;
- Power supply;
- Location of display(s).

The careful top-down integrated approach to post-accident monitoring instrumentation that was requested by the NRC and implemented by IP-2 concluded that one variable, neutron flux instrumentation, 1) was not a Type A variable because at IP-2 it is not required for event identification, event recovery to stabilization, nor maintaining stabilized conditions and recovery to cold shutdown; and 2) need not be upgraded to Category 1 but was acceptable with its current qualification status as Category 3. Based on the latter, Con Edison sought an exception and does not agree with committing to upgrade the neutron flux instrumentation. This report is intended to provide further supporting justification for the exception.

3.0 SAFETY ASSESSMENT

The accidents analyzed in the FSAR for IP-2 where core nuclear power is potentially generated (e.g., main steam line break, boron dilution, etc.) were evaluated to determine the relationship between nuclear power and heat flux. This evaluation found that as core nuclear power increased, core heat flux increased. This is expected since the nuclear fission process itself produces significant heat when a U^{235} atom is split, in addition to the neutrons that sustain the chain reaction, . The increasing core heat flux will increase core exit and Reactor Coolant System (RCS) temperatures given that the capability of the heat removal systems is unchanged or at their maximum. Therefore, core exit and RCS temperature instrumentation do provide a direct means of monitoring the increasing core nuclear power that can result from criticality if it were to occur during these events. Accordingly, new EOP technical guidelines were developed to provide a systematic method to determine the status of the Subcriticality Critical Safety Function during adverse containment conditions. The new EOP technical guidelines are fully described in Section 5.0 of this report.

During adverse containment conditions, the new technical guidelines involve monitoring core and RCS temperature behavior by evaluating core exit temperatures and RCS temperature trending as measured on the core exit thermocouple system and wide range hot and cold leg RTDs. An adequately shutdown core is confirmed after the boron concentration in the containment sump or, as applicable, the RCS is known to be above the minimum shutdown value. Otherwise, boration will continue until a sufficient inventory of borated water is injected. These technical guidelines would be used by an operator as enhancements made to the EOPs.

3.1 Design Basis Accident Analysis Evaluation

A discussion of the design basis accident analysis evaluation including the operator actions based on the flow through the EOPs using the new technical guidelines is provided below:

3.1.1 Loss of Coolant Accident (LOCA)

The fundamental characteristic of the large break LOCA is a rapid depressurization of the RCS and a pressurization of the containment. RCS breaks greater than 2 inches result in the classic LOCA scenario that requires injection of the borated water inventory (at least 2,000 ppm) from the Refueling Water Storage Tank (RWST). The depressurization of the RCS results in a pressure decrease in the pressurizer and a pressure increase in the containment as well as adverse containment conditions. A safety injection actuation signal is generated when the appropriate low pressurizer pressure setpoint is reached. For the large break LOCA these signals occur essentially instantaneously with the break. For smaller breaks this setpoint is reached very quickly since they are typically set only about 350 psi below normal operating pressure. These protective countermeasures limit the consequences of the LOCA in two ways. First, reactor trip and borated water injection complement void formation by causing a rapid power reduction to fission product decay heat levels. Second, the injection of borated water provides for heat transfer from the core, prevents excessive fuel clad temperature, and maintains subcriticality throughout the scenario.

The blowdown of RCS fluid causes pressure, humidity and temperature levels in containment to rise, and, when containment pressure reaches the high-high containment pressure setpoint, chemical spray is injected into the containment atmosphere. This blowdown also causes significant core voiding, core uncovery and reactor vessel downcomer uncovery. During the LOCA, the blowdown and spray fluids mix with the containment air to create an adverse environment in which exposed equipment that is relied on to detect and mitigate the event must function. The reactor trip signal and resultant control rod insertion occur within a very short period from event initiation and is indicated in the control room by several diverse means. Once the reactor is tripped, subcriticality is maintained by the injection of borated water as discussed above. If an operator cannot verify that the reactor has tripped when it is required to be tripped, he is directed by the EOPs to manually trip the reactor. If reactor trip still cannot be verified, EOP E-0, Step 1 directs the operator to enter function recovery procedure FR-S.1 and commence emergency boration of the RCS, which is already in progress due to safety injection. Since the LOCA scenario results in significant core heatup above 700°F, the operators would initially be directed by the new EOP technical guidelines to implement FR-S.1, to determine if sump and RCS boron concentrations are above minimum shutdown values and to monitor core heatup. If the boron concentration is not known to be greater than the minimum shutdown value (no samples have been taken, analysis is not completed, etc.), the operator would continue to carry out FR-S.1 and emergency borate the RCS, thereby assuring the core is subcritical. For the large or intermediate break LOCA, since the reactor is reflooded with borated water from the RWST and the contents of this tank and the borated Emergency Core Cooling System (ECCS) accumulators (at least 2,000 ppm) are emptied prior to switch over to recirculation, there is, by design of the ECCS, a sufficient supply of borated water to maintain the reactor in a subcritical condition. In fact, the ECCS water has enough boron concentration to maintain the core shutdown for break sizes greater than or equal to 3.0 $ft.^2$, without any credit for shutdown provided by the control rods. WCAP-8339, "Westinghouse Emergency Core Cooling System Evaluation Model - Summary", contains the analysis per the requirements of 10CFR50.46(b)(5) to support this ECCS design criteria. Thus, continuing emergency boron injection and recirculation is fully consistent with the LOCA mitigative strategy of the EOPs. Once this accident is successfully mitigated and core exit temperature is below 700°F, recovery proceeds on long term recirculation of borated cooling water. Periodic core exit and RCS temperature monitoring and periodic sampling to determine boron concentration are sufficient actions to assure that any approach to criticality is detected.

In the highly unlikely event that the reactor should return to power in the recovery phase following mitigation of a LOCA and the heat removal systems (ECCS) cannot maintain stable temperature conditions, the new EOP technical guidelines will again direct the operator to function recovery procedure FR-S.1, until such time as boron concentration is confirmed, core heat up terminated and/or stabilized, and potential dilution paths isolated. During the time required for boron concentration sampling activities, the core heatup resulting from the return to power will be detected by the core exit and RCS temperature instrumentation which are appropriately qualified as Category 1.

The core voiding and core uncovery caused by RCS blowdown during LOCA conditions will heavily influence neutron flux instrumentation response. As the ECCS and two-phase RCS mixture is pumped through the downcomer and core, three effects are manifest: (1) less water in the core decreases the intrinsic neutron source reading; (2) decreased fluid density in the downcomer permits more neutrons to leak out to the excore detectors; (3) increased leakage from the core reduces neutron multiplication. The second effect is by far the most dominant as far as excore detector response is concerned and what information the operator sees. Although fewer neutrons remain in the core to help sustain the fission process, many more are able to escape to the neutron detector for measurement. Therefore, in a voided or uncovered core, the neutron flux readings could be misleading and imply a high neutron flux level when, in fact, the core is effectively shutdown. Under these circumstances, core exit temperature would more appropriately monitor the status of the core since there would be no heat generated by nuclear power in a shutdown core. Additionally, it is important to recognize that voiding different regions of the core will have a varying effect on excore detector readings. For example, voiding the center of the core may affect the neutron population in that vicinity, but any change will be "shielded" from the detector by peripheral fuel assemblies and fluid in the downcomer annulus. Conversely, voiding the downcomer region adjacent to the detector will have a major effect since neutrons reaching that region will be able to travel largely unimpeded to the detector. A one-dimensional neutron transport calculation (performed by the Nuclear Safety Analysis Center (NSAC)) suggests that voiding the downcomer annulus will result in a count rate increase by a factor of 400, and is the dominant mechanism by which source and intermediate range neutron detector signals are affected (under these conditions).

These misleading situations were analyzed following the TMI-2 accident because of two factors: (1) the operators sometimes thought that the core was not shutdown due to the high observed neutron flux readings; and (2) to determine the usefulness of neutron flux measurements to measure reactor water level. For IP-2, reactor water level is measured by the Reactor Vessel Level Instrumentation System (RVLIS) and there is no need to cover it further in this report. The neutron flux readings and the shutdown state of the TMI-2 core were the subject of an extensive analysis presented in Appendix "RECRIT" from NSAC-1 Supplement, issued in October 1979, which is reproduced here as Appendix D. This analysis concluded that there was little likelihood of recriticality or conditions approaching recriticality before the TMI-2 core disarray occurred (the typical condition in a LOCA). The core was actually becoming more and more shutdown even though detector count rate increased, which was primarily due to system and downcomer voids. Thus, pursuing another means to diagnose and mitigate core criticality with an adverse containment accident in progress is appropriate and desirable from a safety and man-machine interface standpoint and entirely consistent with NUREG-0737, Supplement 1 criteria to avoid misleading operators. Monitoring of core exit and RCS temperature conditions with the threshold values presented in Section 5.0 of this report is accurate in indicating the status of the core.

Therefore, for the LOCA which yields an adverse containment environment, following the new EOP technical guidelines is the better method to detect an approach to criticality or to determine that the accident has been successfully mitigated. Since temperature instrumentation which was installed and accepted per NUREG-0737, Supplement 1 already exists at IP-2 and is used with the EOPs, installation of Category 1 neutron flux instrumentation will not provide any safety benefit or increased protection.

3.1.2 Main Steam Line Break (MSLB)

Breaks outside containment will not affect the normal containment environment and the existing neutron flux instrumentation will be used to perform its intended function for these events. The MSLB must occur inside containment in order for an adverse containment environment to be created. In that case, the technical guidelines for an adverse containment will apply.

The fundamental characteristic of a MSLB is a rapid cooldown and depressurization of the intact RCS due to the uncontrolled heat removal via the high blowdown steam flow out the break. The steam generator (S/G) blowdown causes a rapid pressure decrease in the faulted S/G, which initiates a reactor trip signal and safety injection actuation. The rapid RCS cooldown causes a positive reactivity insertion due to the negative moderator temperature coefficient and causes a return to power. The reactivity transient is mitigated by the automatic injection of borated water from the RWST and the reactor is quickly made subcritical again. Automatic emergency boration action is provided for this event due to the rapidness of the positive reactivity insertion. By that time, the S/G blowdown into containment is nearly complete, RCS temperature and pressure stabilizes and temperature is controlled by the remaining intact S/G. Core exit temperature will be less than 700°F due to the rapid RCS cooldown and RCS temperature should then stabilize and/or trend to no-load conditions. The safety injection termination criteria are subsequently met when the water level in the pressurizer returns, RCS pressure is stable or increasing, and adequate subcooling margin exists. Auxiliary feedwater is throttled to maintain level in the intact S/G and to control RCS temperature.

The automatic action of safety injection during this accident is to emergency borate the RCS and accommodate RCS inventory shrinkage due to the rapid cooldown. During the predicted return to power, the RCS decay heat is adequately removed by the blowdown of steam from the faulted S/G. Since it is a design basis condition to expect a return to power following a MSLB, automatic protection equipment is provided in the form of emergency boration via the safety injection system and thus, no operator action is required to mitigate this expected initial reactivity transient.

If RCS temperature does not stabilize and/or trend to no-load temperature, but the core temperature is still below 700°F, emergency boration will be initiated by operator action through the new EOP technical guidelines. This is appropriate because the temperature instability could be due to nuclear power being generated in the core.

Given that RCS temperatures stabilize, emergency boration would continue under the new EOP technical guidelines until RCS boron concentration was confirmed to be above minimum shutdown requirements. For the MSLB that occurs inside containment, it is necessary to sample the intact RCS for boron rather than the containment sump because the secondary plant fluid will be condensed in the sump and it does not contain boron nor communicate with the intact RCS. The new EOP technical guidelines appropriately reflect this potential condition.

In the unlikely and unpredicted event that a return to power were to occur from an unknown boron dilution that may be in progress after the MSLB, the core average temperature would increase due to the increase in core heat flux caused by the generation of nuclear power and, with either forced or natural circulation in the RCS, be detected by the core exit and/or RCS temperature (wide range) indication which would trend upwards. The operators would be directed by the EOPs to implement FR-S.1 and initiate emergency boration of the RCS, until RCS temperature stabilizes and RCS boron concentration is known to be above the minimum shutdown value. Thus, the mitigating actions would be the same as in the LOCA cases discussed above and are consistent with the boron dilution analysis mitigative strategies presented in the IP-2 FSAR.

By using core exit temperature and RCS wide range temperature indications, the status of the core is monitored directly and operator action to emergency borate the core is taken. Therefore, installation of Category 1 neutron flux instrumentation will not provide any safety benefit or increased protection.

3.1.3 All Other Design Basis Accidents

The following list of accidents (or groups of accidents) as presented in the IP-2 FSAR were evaluated:

- Feedwater Enthalphy Decrease
- Excessive Load Increase
- Loss of Load
- Loss of RCS Flow/Locked Rotor
- Loss of Main Feedwater
- Uncontrolled Rod Withdrawal
- Startup of Inactive Reactor Coolant Loop
- Rod Ejection
- Steam Generator Tube Rupture (SGTR)
- Inadvertent Boron Dilution

The evaluation was intended to define conditions that may need to be incorporated into the new EOP technical guidelines discussed in Section 5.0 of this report that were not apparent from the evaluation of the primary and secondary pipe ruptures inside containment. The evaluation found that in no case do these analyses predict that an adverse containment environment will result. Therefore, no special considerations related to these events need to be included in the new EOP technical guidelines as the currently installed neutron flux instrumentation will function in a normal containment environment. Also, since the above events do not involve significant core or downcomer voiding or core uncovery, it is acceptable to rely on the existing neutron flux instrumentation to accomplish the mitigative strategies of the EOPs. Thus, installation of Category 1 neutron flux instrumentation will not provide any safety benefit.

3.2 Accidents Beyond Design Basis

In developing the Emergency Response Guidelines (ERGs), the WOG justified a probability cut-off value of 10^{-8} for identifying functional failure sequences for the LOCA, Secondary Line Break, and SGTR events for which no further procedure development was required. The ERGs are the basis for the IP-2 EOPs. The 10^{-8} cut-off probability covers more than 99 percent of the probability of occurrence for a core melt event scenario. Thus, since the ERGs were developed on that basis, the possible event scenarios covered by the IP-2 EOPs go beyond the FSAR design/licensing basis accidents and transients upon which the plant design features are based. There are 115 beyond design basis event scenarios which form the basis of the EOPs.

Many of the EOP event scenarios involve adverse containment conditions which requires the use of new EOP technical guidelines that do not rely upon the neutron flux instrumentation to verify the subcriticality critical safety function. For these beyond design basis event scenarios, a table-top event by event review was performed with consideration of changing core hydraulic conditions. This review also determined if there are any events involving an adverse environment where flux monitoring instrumentation may be meaningful. The analysis performed by WOG for these event scenarios are contained in several WCAP reports (for example, WCAPs - 9600, 9753, 9744) and WOG letters (for example, OG-57, 62, 63, 72, 91, 92). This analysis information was utilized in the table-top review to assess the changing hydraulic conditions during each event scenario.

The initial assessment of the 115 possible event scenarios considered the three major pressurized water reactor plant accident initiators (loss of reactor coolant, loss of secondary coolant, and steam generator tube rupture) and the functions required to mitigate the consequences of these accidents. For each function, functional failures were defined as shown in Table 3-1. The EOPs that provide coverage for each of the event scenarios were considered together with the predicted hydraulic conditions as contained in the above noted WOG analyses. An assessment was made to determine which of the 115 beyond design basis event scenarios would cause the containment to become adverse, and, which were bounded by others in terms of changing core hydraulic conditions and the EOPs that would be used to recover from the event. This screening reduced the number of beyond design basis event scenarios requiring a table-top review. By proceeding in this manner, redundant table-top reviews of event scenarios were eliminated where no new information would be gained relevant to mitigating the effects of core and/or downcomer voiding and/or uncovery. Ultimately, twenty nine (29) event scenarios beyond the standard design basis were identified that create adverse containment conditions using combinations of the various major accident initiators and functional failures defined in Table 3-1. Their probability of occurrence is within the 10^{-8} EOP basis. The step by step EOP usage was then evaluated in a table-top event by event review for each of the 29 beyond design basis events which are listed in Table 3-2.

Since these scenarios are beyond the design basis (but within EOP basis), the accident is very severe with respect to core conditions. The EOPs are designed to prevent core damage and/or melting for these scenarios and, accordingly, utilize nearly all available plant equipment and systems in recovery actions. In order to go beyond the design basis accident probability to a 10^{-8} cutoff probability, the equipment that has to fail to initiate the scenario is largely the same emergency safety equipment that was built into the plant to prevent such severe core conditions from occurring in the first place. Given the occurrence of the beyond design basis events listed in Table 3-2, severe core voiding, core uncovery, downcomer voiding, downcomer uncovery, and/or loss of subcooling occur during the scenarios and continue at varying degrees throughout the events since the safety equipment built into the IP-2 plant to mitigate these effects is assumed to be in a failed state. These changing hydraulics will cause the neutron flux instrumentation readings to be erroneous and potentially misleading to the control room operators during the recovery. Appendix D contains an analysis supporting this conclusion that was performed by NSAC after the TMI-2 accident. However, the same evaluation has shown that core exit temperature and the stability of RCS temperatures provide a unique and direct indication of core power generation as well as adequacy of core cooling action. The threshold temperature values selected to trigger operator actions in the new EOP enhancements are sufficient to provide conservative guidance to take appropriate emergency boration action that enhances the status of the core even for those cases where the reactor remained shutdown. Thus, the installation of Category 1 neutron flux instrumentation will not provide any safety benefit in recovery from accidents beyond the design basis.

3.3 Boration Requirements

The new EOP technical guidelines involve injecting borated water inventory into the RCS. An adequate boron concentration would then confirm that the core is shut down. The required boron concentration versus cycle burnup is calculated (and plotted) on a cycle specific basis for IP-2. These calculations and plots are done to assure compliance with the plant Technical Specifications and are very conservative with respect to core shutdown margin. A typical plot of minimum boron concentration versus cycle burnup is presented in Figure 3-1. The cycle specific plots are readily accessible in the IP-2 control room graphs book, are controlled by administrative and/or operating procedures, are referenced for use in the EOPs, and the operators are familiar with them since they are available for use on a daily basis. The minimum boron concentrations are calculated with sufficient conservatism to provide margin for the sampling error allowance. Therefore, once the operator obtains a boron concentration at or above the value given in the graph, minimum shutdown margin is assured and no further manipulation of the data is necessary.

3.4 Assessment Results

A safety assessment of the design basis and beyond the design basis accident analyses for IP-2 was performed to determine that by developing and using the new EOP technical guidelines, accident diagnosis and plant recovery are successfully accomplished. For the worst case accidents such as LOCA and MSLB, an adverse containment environment would be generated and the operator would follow the new EOP technical guidelines which are presented in Section 5.0 of this report. These do not rely on neutron flux instrumentation, which is unqualified for Category 1, but on direct reading core exit and RCS temperature instrumentation which are qualified. Therefore, there is no technical reason to upgrade the neutron flux instrumentation to Category 1. For other events, the containment is not expected to become adverse and the existing neutron flux instrumentation can be used. For these cases, the Category 3 variable design and qualification is acceptable for accomplishing EOP functions. If, during any of those events, the containment were to become adverse, the operator would follow the new EOP technical guidelines in an adverse containment. Thus, successful diagnosis and accident mitigation can be achieved without the need to upgrade the neutron flux instrumentation.

TABLE 3-1

Functional Failures

Symbol	Function	Definition of Function Failure
EP	Electrical Power	Failure to provide ac power to buses that furnish power to ESFs
RPS	Reactor Protection System	Failure of more than 2 control rod assemblies to insert in core--electrical/mechanical fault
AFWS	Auxiliary Feedwater System	Failure to deliver the equivalent of full flow of one motor-driven AFW pump
SSR-	Secondary Steam Relief	
SD/S/R-VO		Failure to open of all steam generator (condenser) steam dump, safety and relief valves
SD/S/R-VR		Failure to re-close of all steam generator (condenser) steam dump, safety and relief valves
SDC		Failure to operate of all (condenser) steam dump valves
PPC-	Primary Pressure Control	
SPRAY		Failure to deliver spray/auxiliary spray flow from reactor coolant loop cold legs/CVCS
S/R-VO		Failure to open of all pressurizer safety and relief valves
S/R-VR		Failure to re-close of all pressurizer safety and relief valves

TENERA

TABLE 3-1

Functional Failures
(continued)

Symbol	Function	Definition of Function Failure
CVCS	Chemical and Volume Control System	Failure of charging and letdown functions that prevent cooldown of RCS to cold shutdown
ECI	Emergency Coolant Injection	Failure to deliver borated water from at least 3 accumulators or 1 LHSI pump to RCS cold legs (large LOCA) or initial failure of both trains of HHSI or Failure to deliver flow from at least 1 train of HHSI system
ECR	Emergency Coolant Recirculation	Failure to re-align to cold leg recirculation to inject water into RCS, or failure to re-align to hot leg recirculation
ECI TERM	Emergency Coolant Injection Termination	Operator actions and/or equipment failures that prevent termination of flow from HHSI pumps or failure to align valves for normal charging and letdown via CVCS
MSI	Main Steam Isolation	Failure to isolate main steam line to faulted (or ruptured) steam generator or failure to terminate auxiliary feedwater to that steam generator
RHRS	Residual Heat	Failure to deliver water to RCS cold leg by at least 1 train of RHRS

TABLE 3-2

Beyond Design Basis Event Scenarios

Scenario No.	Initiation	Failed Functions
1	Large Break LOCA	ECR
2	Large Break LOCA	ECI
3	Large Break LOCA	EP
4	Small Break LOCA	SSR-SDC and SSR-SD/S/R-VR
5	Small Break LOCA	SSR-SDC, SSR-SD/S/R-VR, and ECR
6	Small Break LOCA	SSR-SDC, SSR-SD/S/R-VR, and ECI
7	Small Break LOCA	SSR-SDC, SSR-SD/S/R-VO, PPC-S/R-VR, and ECR
8	Small Break LOCA	SSR-SDC, SSR-SD/S/R-VO, PPC-S/R-VR, and ECI
9	Small Break LOCA	SSR-SDC, SSR-SD/S/R-VO, and PPC-S/R-VO
10	Small Break LOCA	AFWS, PPC-S/R-VR, and ECR
11	Small Break LOCA	AFWS, PPC-S/R-VR, and ECI
12	Small Break LOCA	AFWS and PPC-S/R-VO
13	Small Break LOCA	RPS
14	Small Break LOCA	EP
15	Secondary Break	SSR-SD/S/R-VR, PPC-S/R-VR, and ECR
16	Secondary Break	SSR-SD/S/R-VR, PPC-S/R-VR, and ECI
17	Secondary Break	SSR-SD/S/R-VO and ECR
18	Secondary Break	SSR-SD/S/R-VO and ECI
19	Secondary Break	AFWS, PPC-S/R-VR, and ECR

TABLE 3-2

Beyond Design Basis Event Scenarios
(continued)

Scenario No.	Initiation	Failed Functions
20	Secondary Break	AFWS, PPC-S/R-VR, and ECI
21	Secondary Break	AFWS and PPC-S/R-VO
22	Secondary Break	AFWS and SSR-SD/S/R-VO
23	Secondary Break	MSI, AFWS, PPC-S/R-VR and ECR
24	Secondary Break	MSI, AFWS, PPC-S/R-VR and ECI
25	Secondary Break	EP
26	SGTR	ECI TERM, PPC-S/R-VR, PPC-SPRAY, and MSI (Ruptured S/G)
27	SGTR	ECI TERM, PPC-S/R-VR, PPC-SPRAY, SSR-SD/S/R-VO, and MSI (Ruptured S/G)
28	SGTR	ECI TERM, PPC-S/R-VR and AFWS
29	SGTR	RPS

Typical Graph of Minimum Shutdown Boron

Concentration Versus Cycle Burnup,

No Xenon Conditions

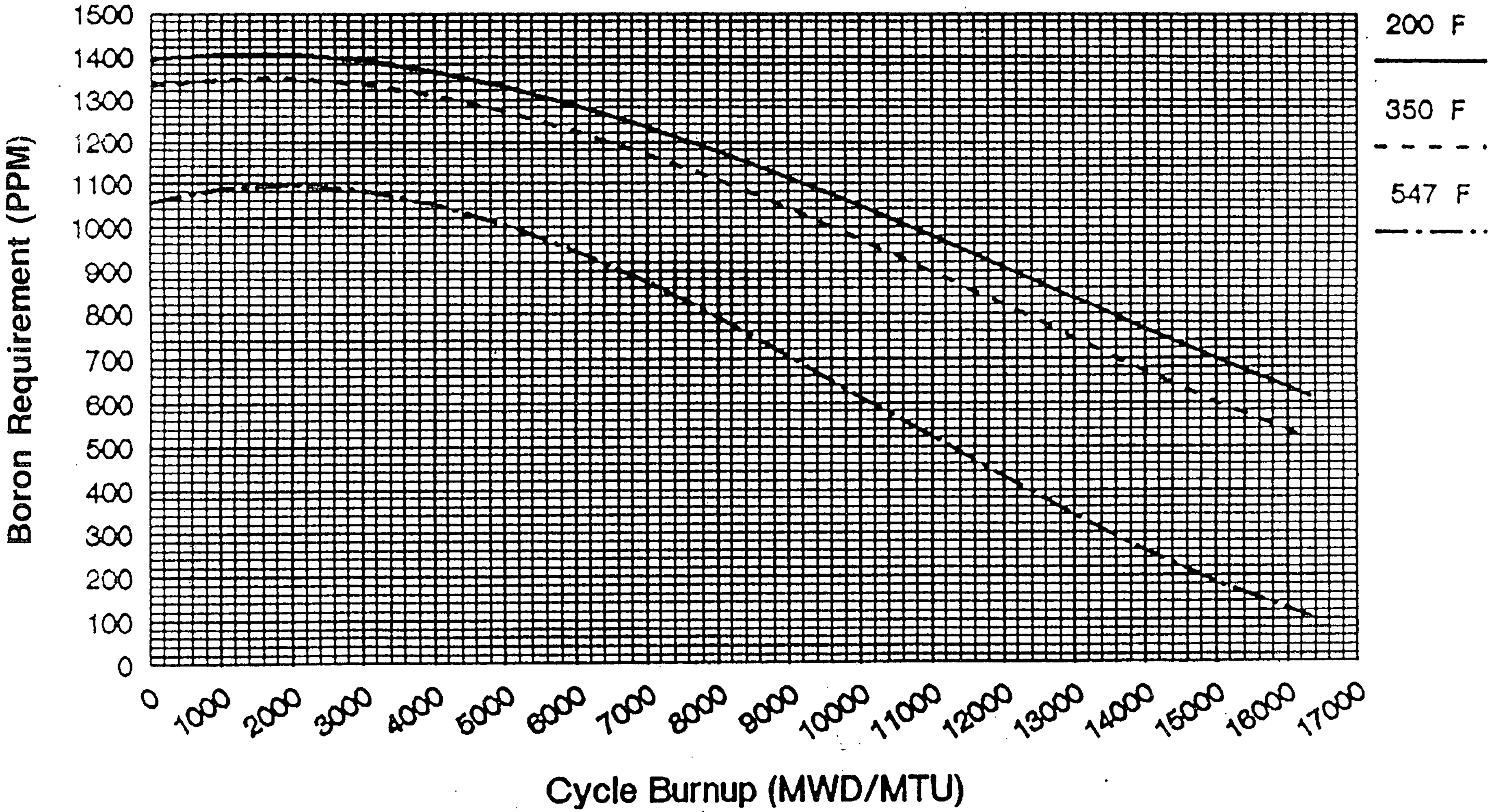

FIGURE 3-1

4.0 COST-BENEFIT ANALYSIS

Having determined that no safety benefit or increased protection will result from upgrading the neutron flux instrumentation in IP-2, even for accidents beyond the design basis event defined paths, and having developed EOP technical guidelines for use when an adverse containment exists that do not rely on neutron flux instrumentation, there is no technical reason to upgrade the instrumentation to Category 1. Further, since other non-adverse containment accidents can utilize the existing neutron flux instrumentation, there is no need for upgrading these instruments for those events. In addition, the following shows that excessive expenditures would be encountered by Con Edison in order to make this unnecessary upgrade.

4.1 Approach and Method

A cost analysis was performed to estimate all costs associated with this potential plant upgrade. This analysis shows that there is little justification to offset the high cost of upgrading the plant.

The actions taken by IP-2 in response to adopting a regulatory retrofit item comprise a functional response approach which was the method used to develop this cost analysis. This approach is based on the method outlined in NUREG/CR-3971 entitled, "A Handbook for Cost Estimating." Cost estimates (direct and indirect) were obtained for the various cost elements associated with each applicable functional response item. A listing of the functional response items and the cost elements considered in this analysis are provided in Tables 4-1 and 4-2 respectively.

The cost of installing a neutron flux monitoring system is dependent upon numerous variables. These variables range from plant configuration, availability and access to existing equipment, cable trays, availability of containment penetrations, etc. and the objective for the system. The objective of an upgraded neutron flux monitoring system is primarily driven by Revision 2 to Regulatory Guide 1.97, Appendix R to 10CFR50 and individual utility requirements. Figure 4-1 illustrates a typical upgraded neutron flux monitoring system design. This cost estimate assumes that the minimum upgrade is being installed which consists of a dual train neutron flux monitoring system that is fit into existing panels and racks. Additionally, this estimate is based on installing a seismically and environmentally qualified system to fully meet Category 1 requirements. The Results Table below provides an order of magnitude cost estimate for installing upgraded neutron flux monitoring system hardware within the existing instrument racks, panels and cabinets in IP-2. Following the table is a brief description of the factors that went into determining the cost for each item identified in the table.

4.2 Results Table

Cost Estimate For Upgrading Neutron Flux Monitoring Instrumentation

Description	Price (1991 Dollars)
o Detector Assembly and Associated Equipment (Materials only)	$1,139,000
o Electrical Penetration Assembly Upgrade (Materials and Labor)	$719,000
o Installation Labor, Support Equipment and Materials (Labor and Miscellaneous Materials)	$947,000
o Equipment Removal, Storage and Radwaste Disposal (Materials and Labor)	$335,000
TOTAL (Minimum Upgrade)*	$3,140,000

Note: *Accumulated Funds Used During Construction (AFUDC) are in excess of these costs.

4.3 Results Table Discussion

Each bullet item that appears in the Results Table is discussed below.

- Detector Assembly and Associated Equipment:

  A typical detector assembly qualified under IEEE performance requirements performs a dual function in that both source and intermediate range monitoring capability is provided, typically called the wide range. The environmentally qualified system also functions during normal operation as well as accident conditions. The detector assemblies and support components are vendor supplied. The estimate includes:

	Item Description	Quantity
a.	Source and Intermediate Range Assembly	3 (includes one spare)
b.	Amplifier	3 (includes one spare)
c.	Signal Processor	3 (includes one spare)
d.	Cable Assemblies (up to inside containment junction box)	2
e.	Indicators (4 per train, 2 trains in plant and 2 trains in simulator)	16
f.	2 pen recorder (one for plant, one for simulator)	2
g.	I/I Converter	2
h.	Transfer switch (1E Qualified)	1 (for Appendix R interface)

- Electrical Penetration Assembly Upgrade:

  Two electrical penetrations will be required, one per train. Existing penetrations within the plant will be upgraded to support the installation of the redundant Category 1 qualified system. The penetration shall be designed, fabricated, installed and tested per ASME Boiler and Pressure Vessel Code Section III, Division 1, subsection NE for class MC Vessels. Appropriate IEEE nuclear standards will be invoked to require seismically and environmentally qualified penetrations. The estimate provided in the table includes materials and labor to upgrade the penetrations.

- Installation Labor and Materials:

  The labor to design, engineer, install, test and document this upgrade involves the cost of labor for the items listed in Table 3-4 to varying degrees.

  In estimating the cost of installing a neutron flux monitoring system various categories had to be addressed, such as engineering, design, material procurement, installation and testing. The design hours include engineering, design package preparation, safety evaluations, safety reviews and approvals, committee reviews, updating of files and licensing documents. The total estimated utility hours for this phase of the project is 4,000 manhours.

  The installation phase of this estimate includes utility managed project planning, trade/craft time doing the actual installation, health physics support, ALARA concerns, decontamination, QA/QC support. Additional testing of the system at various stages of installation and ensuing operability testing of the completed system will also be performed. The estimated utility hours for this phase is 5,000 manhours.

The cost estimate for the miscellaneous materials and equipment needed for this installation effort include:

a. Scaffold installation and removal
b. Temporary lighting
c. Conduit & supports
d. Cable tray & supports
e. Pull boxes & connectors
f. Radiant energy shield
g. Instrument cable (outside containment to control panel)
h. Class 1E power feed
i. Removal of old control panel and miscellaneous field equipment
j. Temporary lifting rig and its support equipment
k. Temporary shielding
l. Radiation protection clothing and devices
m. Other miscellaneous installation equipment (unistrut, mounting hardware, tools, brackets, etc.)

This excludes the labor costs for the electrical penetration assemblies upgrade.

- <u>Equipment Removal, Storage and Radwaste Disposal</u>:

Existing neutron flux detectors, cables, penetration assemblies, support equipment and installation materials will have to be removed from the plant as they will not be used for the new system and are contaminated. This estimate includes the removal of the old system and the trade and staff labor and materials for fabrication of a lead casket for transporting and storing the contaminated expendable detectors and electrical penetration assemblies. Also included is the cost of radwaste disposal (labor, materials, transportation, etc.) via a LSA cask.

4.4 Conclusion

The minimum order of magnitude cost to upgrade neutron flux instrumentation to meet Category 1 (R. G. 1.97, Rev. 2) criteria for IP-2 is estimated to be $3,140,000. Based on this technical and regulatory evaluation, ***an approximate minimum cost of over $3 million for the IP-2 plant to upgrade is excessive when little or no safety benefit can be derived.***

TABLE 4-1

Functional Response Items Considered

1. Analyze the regulatory retrofit item
2. Meet with NRC
3. Prepare responses to NRC
4. Answer questions from NRC Inspectors and verbal communication with headquarters
5. Perform conceptual design, including unresolved safety question determination, resource estimate, and preliminary schedule.
6. Evaluate budget requirements
7. Perform detailed design and/or design review, including specifications for outside procurement.
8. Perform safety/risk/reliability analysis
9. Procure materials and equipment, including preparation of the bid package, evaluation of proposals, and preparation of purchase order.
10. Plan installation, including detailed procedures, labor requirements, schedule installation equipment, temporary facilities, etc.
11. Modify structures
12. Install, test and maintain hardware
13. Inspect hardware
14. Develop software
15. Add to or change record keeping
16. Write/rewrite procedures
17. Conduct test of system/subsystem
18. Write/rewrite training manuals
19. Train/retrain staff
20. Review Technical Specifications and FSAR
21. Modify structures in a radiation environment
22. Install, test and maintain hardware in a radiation environment
23. Draft license amendment

TABLE 4-2
Cost Elements Considered

1. Project Management Labor
2. Engineering Labor
3. Clerical Labor
4. Drafting Labor
5. Programming Labor (Simulator, SPDS, Plant Computer, NIS Rack)
6. Administrative Labor
7. Accounting Labor
8. Quality Assurance/Quality Control Labor
9. Executive Labor
10. Craft Supervisory Labor
11. Craft Labor
12. Radiation Protection Labor
13. Security Labor
14. Technician Labor
15. Computer Usage
16. Equipment (New System and associated installation equipment)
17. Materials (New System and associated installation materials)
18. Simulator (Hardware and Software)
19. Reproduction
20. Storage of Contaminated Equipment (Old system removal and any installation equipment)
21. Accumulated Funds Used During Construction (AFUDC)

NEUTRON FLUX MONITORING SYSTEM WITH APPENDIX R INTERFACE

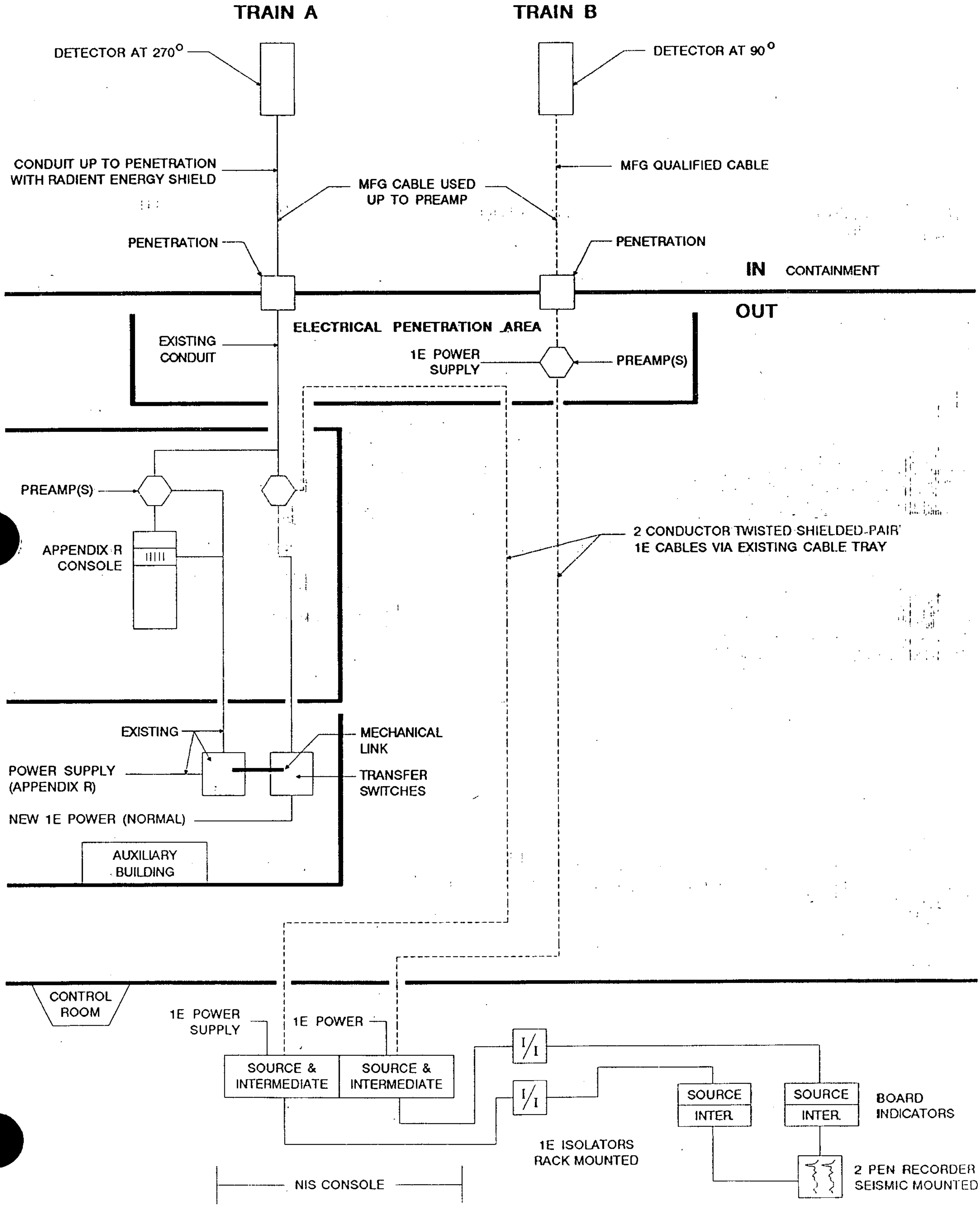

FIGURE 4-1

5.0 BASIS FOR EOP TECHNICAL GUIDELINES

5.1 Subcriticality Criteria for EOP Technical Guidelines

The EOPs provide a systematic method to explicitly determine the status of the Subcriticality Critical Safety Function during adverse and normal containment conditions.

During normal containment conditions, the existing F-0.1 status tree is monitoring the reactivity state of the core by evaluating the parameters characterizing neutron (leakage) flux behavior as measured by the excore nuclear instrumentation system (NIS). An adequately shutdown core typically exhibits below measurable activity on the power range and intermediate range and a randomly fluctuating count rate on the source range instruments. For the purpose of the status tree, the core is considered adequately shutdown (subcriticality satisfied) whenever the level of shutdown is steady or decreasing in the source range (zero or negative startup rate). The F-0.1 tree represents the highest priority Critical Safety Function and, as such, is always entered first any time tree monitoring is initiated. The tree directs operators to either of two Function Restoration Procedures.

During adverse containment conditions, the new proposed EOP technical guidelines monitor core and RCS temperature behavior by evaluating core exit temperatures and RCS temperature trending as measured on the qualified core exit thermocouple system and wide range hot and cold leg RTDs. An adequately shutdown core is confirmed when the nuclear heat generated by a critical core is below threshold temperature values. Additionally, the boron concentration in the containment sump or, as applicable, the RCS, is monitored to determine if it is known to be above the minimum shutdown value. Otherwise, boration will continue until the appropriate parameters are met.

The basis of each of the EOP technical guidelines are discussed below. These guidelines are:

- o Core exit thermocouples less than 700°F.
- o Containment sump boron concentration know to be greater than minimum shutdown value.
- o RCS temperature stable and/or trending to no load T-average.
- o RCS boron concentration known to be greater than minimum shutdown value.

These would be used by an operator as enhancements made to the EOPs.

PROPOSED EOP TECHNICAL GUIDELINES

EOP GUIDELINE: Core Exit Thermocouples Less Than 700°F

PURPOSE: To determine if the core heat flux being generated by significant nuclear power is sufficient to raise the core exit temperature above a value where the maximum plant decay heat removal capability is insufficient.

BASIS: Following a reactor trip, nuclear power and core heat flux promptly drop to only a few percent of nominal, and then decay away. Decay heat levels resulting from radioactive fission product decay are never more than a few percent of nominal power and also decrease in time with a steady decrease in core heat flux. At a constant heat removal rate, core exit and RCS temperatures should remain stable and trending to no-load conditions, and as decay heat levels decrease the heat removal capacity is sufficient to reduce core and RCS temperatures in a controlled manner. During a LOCA, the coolant is depleted and core temperatures increase above nominal values. The ECCS design capability automatically reverses the core temperature increase by injecting colder borated emergency coolant from the RWST. Should post-LOCA core temperatures unexpectedly increase again*, this may be indicative that core nuclear power is being generated and emergency boration should commence.

Note: * A boron dilution event occurring after LOCA or MSLB recovery are completed and the plant has been stabilized will yield an increase in core nuclear power and corresponding increase in core heat flux. Should this exceed the heat removal capability of the ECCS, core exit temperature will exceed 700°F.

During a MSLB, the coolant is rapidly cooled and core temperatures decrease sharply and significantly below 700°F until the blowing down S/G boils dry. The sharp decrease in temperature causes a rapid positive reactivity insertion in the core which is automatically reversed by the ECCS injection of borated emergency coolant. Once the faulted S/G is boiled dry, core temperature should begin to stabilize and trend to no-load average temperature conditions, since the intact S/Gs will once again control RCS and core temperatures. Should post-MSLB core temperatures unexpectedly increase*, this may be indicative that core nuclear power is being generated and emergency boration should commence.

Safeguards heat removal systems are sized to remove only decay heat and not significant core nuclear power which will cause core and RCS temperatures to be unstable and increasing. The 700°F is chosen because generic Westinghouse analyses that form the basis of the ERGs indicate that operator-initiated recovery actions are needed to respond to a core condition where maximum design decay heat removal is not able to match core heat generation. When core exit temperature exceeds approximately 700°F, degraded core conditions can exist and operator action to terminate the heat generation (i.e., emergency boration) should be initiated unless the boron concentration is known to be above

Note: * A boron dilution event occurring after LOCA or MSLB recovery are completed and the plant has been stabilized will yield an increase in core nuclear power and corresponding increase in core heat flux. Should this exceed the heat removal capability of the ECCS, core exit temperature will exceed 700°F.

the minimum shutdown value. Once emergency boration is initiated, the operator can implement subsequent steps and carry out other function restoration procedures as directed by the status trees.

INSTRUMENTATION: Core Exit Thermocouples

PROPOSED EOP TECHNICAL GUIDELINES

EOP GUIDELINE: Containment Sump Boron Concentration Known To Be Greater Than Minimum Shutdown Value

PURPOSE: To confirm that emergency boration recovery action can be terminated or is not necessary.

BASIS: As the EOPs are followed and core exit temperature has been determined to be indicative of a potentially degraded core condition (i.e. about 700°F) a severe challenge to the Subcriticality Critical Safety Function may exist and core shutdown status needs to be confirmed. Because RCS pipe ruptures cause primary coolant to spill into containment, this will fill the sump. As LOCA recovery progresses, highly borated water (ECCS) is injected into the RCS from the RWST and ECCS accumulators and mixes with the spilled water in the containment sump. Subsequent to automatic ECCS injection, switch-over to sump recirculation is made. A containment sump boron sample would contain the boron concentration of the recirculating ECCS fluid in the core. Therefore, if the containment sump boron concentration is not known, or the boron sample analysis results show a boron concentration below the minimum shutdown value for that time in fuel cycle life, then the excessive heat indicated by the core exit T/Cs (>700°F) is due to nuclear power generation in excess of the heat removal capability and the core must be shutdown.

For a secondary plant pipe rupture inside containment the containment sump would not be expected to have a boron concentration greater than the minimum shutdown value since the secondary plant fluid is not borated.

<u>INSTRUMENTATION:</u> Boron analysis from the Post-Accident Sampling System.
Control room graphs of minimum shutdown margin versus cycle burnup.

PROPOSED EOP TECHNICAL GUIDELINES

EOP GUIDELINE: RCS Temperature Stable and/or Trending to No Load T-AVE

PURPOSE: To determine if emergency boration may be needed for a slowly developing nuclear power generation transient even though core exit temperatures are within acceptable limits and core heat addition is balanced with RCS heat removal capability.

BASIS: Given that core exit temperature is less than 700°F, this guideline will be used to decide if further evaluations should be directed at determining if the RCS boron concentration is above the minimum shutdown value. After a reactor trip, RCS temperature should stabilize and/or be trending to no-load temperature values. Subsequent to a MSLB event, after the rapid cooldown of the RCS is terminated and the reactivity insertion is automatically reversed by the ECCS, the plant conditions should again stabilize and/or trend to no-load temperatures.

RCS temperature stable and/or trending to the no-load value indicates that the core heat input is balanced with the capability of the heat removal systems, as designed. If RCS cooldown is excessive due to excessive feed to the steam generators following a main steam line rupture, this can also result in continuing to cool down the RCS and it may be necessary to initiate emergency boration to prevent generation of nuclear power.

If RCS temperature is greater than no-load values and increasing when the decay heat removal systems are at their maximum, then emergency boration is required because the heat input is not balanced.

If RCS temperature is stable and trending to no-load conditions, then the operator is directed to confirm the RCS boron concentrations are adequate. Since a possible event that causes the cooldown may have been a MSLB inside containment, the sump will be filled with non-borated secondary plant water. Thus, the RCS sample will be necessary.

INSTRUMENTATION: RCS Hot and Cold leg wide range RTDs.

PROPOSED EOP TECHNICAL GUIDELINES

EOP GUIDELINE: RCS Boron Concentration Known to be Greater Than Minimum Shutdown Value

PURPOSE: To confirm that emergency boration recovery action can be terminated or is not necessary based on the boron concentration in the core.

BASIS: As the EOPs are followed and core exit temperature is greater than 700°F and the containment sump boron concentration is greater than the minimum shutdown value, additional confirmation of shutdown margin is established by this guideline. This would be a typical point in a post-LOCA recirculation scenario and this is a final check that the boron concentration in the core is sufficient to keep the core from generating significant nuclear power. This accommodates any possible difference in sump and RCS boron concentrations. Upon obtaining that confirmation, the operator can be confident that any high temperature is not due to nuclear power generation and that there is no unexpected boron dilution event also in progress. During the post-LOCA recovery the sampling of the sump and RCS will be initiated periodically, so the status of the core will continue to be reaffirmed.

Should the boron concentration in the sump be above the minimum shutdown value but for some unknown reason the boron concentration in the RCS is not, emergency boration is required.

Proceeding in this manner assures a conservative response since a high core temperature could be due to nuclear power generation that may be occurring from an unexpected boron dilution of the

RCS during post-LOCA recovery via branch RCS loop connections.

The RCS boron concentration, when plant recovery is at a point where core exit temperature is below 700°F and the RCS temperature is stable and trending to no-load T-AVE conditions, would also be established. This could be a typical point in a post-MSLB inside containment scenario and this becomes a final check that RCS boron concentration is sufficient to prevent nuclear power generation. By initiating this RCS sample periodically, the operator is assured that no unexpected post-MSLB boron dilution is occurring.

If boron concentration in the RCS is found to be below the minimum shutdown value and low core exit temperature and stable RCS temperature at no-load exists, this is considered to be a potential loss of shutdown margin and a challenge to the Critical Safety Function may exist.

By conservatively proceeding in this manner, the operator is assured that any slowly developing boron dilution in a post-LOCA or post-MSLB is diagnosed and mitigated before nuclear power generation causes core and RCS temperatures to significantly increase. In an adverse containment, only when core exit temperature is below 700°F, RCS temperature is stable and at the no-load value and RCS boron concentration is confirmed to be above the minimum shutdown value is the Subcriticality Critical Safety Function satisfied.

INSTRUMENTATION: Boron analysis from the Post-Accident Sampling System.
Control room graphs of minimum shutdown margin versus cycle burnup.
RCS Hot and Cold leg wide range RTDs.

5.2 Assessment of EOP Technical Guidelines

Revision 1A of the Westinghouse Owners Group Emergency Response Guidelines (Rev. 1A-ERGs) is the basis for the Emergency Operating Procedures (EOPs) in the IP-2 plant. The functional capabilities of plant systems and components relevant to EOPs have been compared to those of the generic reference plant. The EOPs were generated by changing the generic guidelines to address differences in functional capabilities and operating characteristics of plant equipment, control room design, operator knowledge requirements, and plant instrumentation. Plant specific EOPs and EOP revisions are acceptable as long as the differences from the generic guidelines are not safety significant. Thus, the new EOP technical guidelines were further reviewed to determine if the differences from Rev. 1A-ERGs are safety significant. This review verified that the IP-2 plant remains within the technical basis of the generic guidelines of the Rev. 1A-ERGs and that the new EOP technical guidelines are fully consistent with the EOP diagnosis and mitigative strategies and that no safety significant deviation exists. ***In fact, this review determined that the generic status tree F-0.1 in Rev. 1A-ERGs should be used with much caution during design basis and beyond design basis events involving core voiding or core uncovery situations because the use of neutron flux indications for determining subcriticality in these situations can be misleading. Hence, the generation of EOP technical guidelines for use in an adverse containment that do not rely on neutron flux indication to determine subcriticality is appropriate.***

The general criteria used for identifying whether a plant specific difference is a safety significant deviation are presented in Standard Review Plan 13.5.2 and include:

a. any modification to the mitigative strategy of the generic technical guidelines.

b. differences in equipment operating characteristics, such as RCP trip criteria and SI termination criteria.

c. differences in equipment operating characteristics, such as SI pumps that can be throttled versus only on/off.

d. identification of methods and equipment used to address the technical areas of the generic guidelines that are specified as plant specific.

e. plant-specific setpoints or action levels that are calculated or determined in a manner other than specified in the generic technical guidelines.

f. actions that are taken in addition to those specified in the generic guidelines and that affect the mitigative strategy.

The new EOP technical guidelines are for use in an adverse containment condition only. The instrumentation used includes:

a) core exit thermocouples;
b) RCS hot and cold leg wide range RTDs; and
c) post-accident sampling system boron analysis.

This equipment has been designed and/or upgraded to meet NUREG-0737 and Supplement 1 requirements and is already part of the post-accident monitoring capability for IP-2. Chemistry personnel are required to periodically demonstrate their familiarity with the Post-Accident Sampling System (PASS) equipment. The technical guidelines are appropriately designed so that they conservatively direct operator action to perform EOP FR-S.1 based on direct reading of core temperature and use the knowledge of boron concentration (from the PASS) to confirm the need to continue FR-S.1 or terminate it. As such, the criteria are fully consistent with the ERG mitigative strategies and IP-2 remains within the technical basis of the generic Rev. 1A-ERGs. Therefore, their use in the EOP set does not constitute a strategic difference from the generic Rev. 1A guidelines.

The generic guidelines and background documents were then evaluated to determine the strategic safety considerations addressed by each guideline and the impact these may have on the usage of the new EOP technical guidelines in an adverse containment. This included the sequence of recovery actions critical to the success of the recovery process and information relating to the structure and interaction of the procedures. The review verified that those EOPs which address an anticipated challenge to plant safety are not adversely impacted and are fully consistent with the approach taken. In fact, core and RCS temperatures and sampling are used throughout the EOP set at various threshold values for assessing plant status and success of varying mitigative actions. The values of these parameters discussed above are based on a sound engineering evaluation of the effects of reactivity insertions due to boron dilution or continued nuclear power generation due to reactor trip anomalies and uncontrolled RCS cooldown during design basis and beyond design basis events. The resulting guidance does establish that rapid emergency boration be initiated in a conservative manner. Use of boron analysis by sampling is also used in many EOPs such as ES-0.2, ES-1.2, ES-1.3, ES-3.1, ES-3.2, ES-3.3, ECA-0.1, ECA-3.1, ECA-3.2 and ECA-3.3. In each case, the specified mitigative action will continue until the sample is drawn and analysis confirms that they are not necessary.

Therefore, it is acceptable to establish EOP technical guidelines for adverse containment conditions that do not rely on neutron flux instrumentation for use in the EOP set.

6.0 POST-ACCIDENT SAMPLING SYSTEM REVIEW

In accordance with the requirements of NUREG-0737, Item II.B.3, the Post-Accident Sampling System (PASS) installed at IP-2 is designed to provide analysis of reactor coolant and the containment during normal operating and post-accident conditions. As a part of this effort, the PASS design was reviewed and the procedures governing its use were walked down to ensure the capability of obtaining a reactor coolant sample for the purpose of determining boron concentration.

Samples of reactor coolant are analyzed by either of two methods: (a) an in-line boron analyzer that will automatically obtain and analyze the sample; or b) manually obtaining a "grab" sample and transporting the sample to the radiological chemistry laboratory for analysis of boron concentration. Samples of the reactor coolant may be obtained from various sample points (hot legs, containment recirculation sump, RHR system).

In order to ensure the capability of obtaining and analyzing a reactor coolant sample to determine the boron concentration, various parameters were reviewed during the PASS review and plant walkdown. The parameters reviewed were:

- Personnel available
- Access and egress routes to PASS control panels and sample stations
- Access and availability to chemistry laboratory and support equipment
- Communications between control room and watch chemistry technician
- Lighting (normal and emergency)
- Time to obtain sample
- Time to analyze sample (automatically and manually)
- Availability of PASS equipment
- System design criteria
- Operating procedures and practices
- Maintenance history of PASS equipment and support systems

- o Reliability of PASS
- o Surveillance requirements

The PASS review concluded that the availability and reliability of the PASS and the required support systems are acceptable and will enable plant personnel to obtain and analyze a reactor coolant system sample to ascertain the boron concentration during normal and post-accident operating conditions for IP-2.

7.0 EMERGENCY BORATION

7.1 Control Room Walkthrough

The emergency boration system was assessed to determine whether the emergency boration capabilities can function as designed to carry out the EOPs. Control room improvements have previously been completed in the plant to enhance the commencement of emergency boration and facilitate the usage of controls by the operators. Human factors improvements such as better instrument scales and their placement, rearrangement of controls, clear labeling of alarms, etc. remain valid. Since no new emergency boration performance requirements are introduced by the new EOP technical guidelines and since the Control Room Design Review already included the emergency boration system, it is concluded that the reactor operators can easily and effectively emergency borate the plant.

7.2 Availability and Reliability Assessment

The emergency boration system is included in the Technical Specifications. The limiting conditions for operation and the surveillance requirements were evaluated and judged to be appropriate to assure continued availability and reliability of emergency boration capability. Needed maintenance is completed to assure operability of the system and sub-systems, otherwise the plant is brought to a shutdown condition.

Therefore, it is concluded that operations personnel can confidently access controls and initiate emergency boration at IP-2 and no new requirements need be imposed on the system as a result of the new EOP technical guideline usage.

8.0 CONCLUSIONS

Con Edison has determined that NUREG-0737, Supplement 1 requirements are met without the need to upgrade the neutron flux instrumentation to Category 1 and has accordingly completed this supplemental evaluation to provide further technical justification for the NRC to grant an exception to the recommendations of Regulatory Guide 1.97, Revision 2. The safety assessment has shown that Emergency Operating Procedures can be revised to enable the critical safety function of subcriticality to be monitored directly by other qualified instrumentation without reliance on neutron flux measurements. Further, existing post-accident sampling and emergency boration systems are capable of supporting this approach. Also, this evaluation has shown that under certain thermohydraulic core conditions which create an adverse containment environment that mandates Category 1 qualification requirements, neutron flux readings can be misunderstood by an operator which is contrary to the objective of NUREG-0737, Supplement 1. Therefore, providing an instrument that is qualified to function in an adverse containment environment would not ensure that the instrument readings are accurate. Finally, an order of magnitude cost estimate shows that approximately $3.14 million would be required to upgrade the neutron flux instrumentation. This cost is excessive especially when little or no safety benefit nor increased protection can be derived by such an upgrade. Therefore, to be fully responsive to the NUREG-0737, Supplement 1 integrated man-machine interface assessment strategy, no safety benefit will be gained by upgrade of the neutron flux instrumentation to the guidance of Regulatory Guide 1.97, Revision 2 and the requested exception should be granted.

9.0 REFERENCES

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6. Indian Point 2 Graph RCS-4 from Graphs Book.

7. Con Edison letter, J. D. O'Toole to D. Eisenhut (NRC), "Response to NUREG-0737, Supplement 1," April 15, 1983.

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22. Con Edison letter, J. D. O'Toole to Director of Nuclear Reactor Regulation (NRC), Core Damage Assessment Methodology, October 29, 1984.

23. Indian Point 2, Chemistry Procedures IPC-I-013, Rev. No. 3, dated 7/28/86 and IPC-E-001, Rev. No. 7, dated 6/21/89, System Operating Procedure 3.2, Rev. 3, dated 9/28/84, Abnormal Operating Instruction 3.4, Rev. 3, dated 2/9/90.

24. Con Edison Drawing No. A227178-09, Sampling System Flow Diagram.

25. NRC letter to Licensees of Operating Plants, et. al., July 2, 1982, CRGR Charter.

26. NRC Manual Chapter 0514, NRC Program for Management of Plant Specific Backfitting of Nuclear Power Plants, August 1988.

27. NUREG-0737, Clarification of TMI Action Plan.

28. U.S. General Accounting Office Report GAO/RCED-86-27, Process for Backfitting Changes in Nuclear Plants Has Improved, December 1985.

29. Report EGG-M-14082, Monitoring PWR Reactor Vessel Liquid Level with SPND During LOCAs, September 1982.

30. Indian Point 2 Emergency Operating Procedures.

31. Title 10 Code of Federal Regulations.

APPENDIX A

SUPPLEMENT 1 REFERENCES TO GUIDANCE DOCUMENTS

APPENDIX A

This Appendix contains some relevant quotations in pertinent part from Supplement 1 stressing that guidance documents not be used as requirements.

Page 1, cover letter -

"The enclosures to this letter are a distillation of the basic requirements for these topics from the broad range of guidance documents that the NRC has issued (principally NUREG report and Regulatory Guides). It is our intent that the guidance documents themselves, referred to in the enclosures, are not to be used as requirements, but rather that they are to be used as sources of guidance for NRC reviewers and licensees regarding acceptable means for meeting the basic requirements."

"You should also note that the staffing levels in table 2 to the enclosure are only goals, and are not strict requirements."

Enclosure, Page 1 -

"..........It is not intended that these guidance documents (NUREG reports and Regulatory Guides) be implemented as written; rather, they should be regarded as useful sources of guidance for licensees and NRC staff regarding acceptable means for meeting the fundamental requirements contained in this document. It is also not intended that either the guidance documents or the fundamental requirements are to be considered binding legal requirements at this time.........."

Enclosure, Page 2 -

"...........The Commission does not believe that existing guidance should be imposed in this manner, but rather that it be used as guidance to be considered in upgrading emergency response capabilities. This indicates the distinction which the staff believes should be made between the requirements and guidance."

Enclosure, Page 3 -

"2. Use of Existing Documentation

The following NUREG documents are intended to be used as sources of guidance and information, and the Regulatory Guides are to be considered as guidance or as an acceptable approach to meeting formal requirements. The items by virtue of their inclusion in these documents shall not be misconstrued as requirements to be levied on licensees or as inflexible criteria to be used by NRC staff reviewers..........

1.97 - Instrumentation for Light-Water Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident.........."

Enclosure, Page 5 -

"..........Any regulatory position that would require the removal or major modification of existing emergency response facilities or equipment requires the specific approval of the responsible Office Director."

Enclosure, Page 13 -

"6. REGULATORY GUIDE 1.97 - APPLICATION TO EMERGENCY RESPONSE FACILITIES

6.1 Requirements

a. Functional Statement

Regulatory Guide 1.97 provides data to assist control room operators in preventing and mitigating the consequences of reactor accidents.

b. Control Room

Provide measurements and indication of Type A, B, C, D, E variables listed in Regulatory Guide 1.97 (Rev. 2). Individual licensees may take exceptions based on plant-specific design features.........."

Enclosure, Page 14 -

"..........Staff review will be in the form of an audit that will include a review of the licensee's method of implementing Regulatory Guide 1.97 (Rev. 2) guidance and the licensee's supporting technical justification of any proposed alternatives..........

Deviations from the guidance in Regulatory Guide 1.97 (Rev. 2) should be explicitly shown, and supporting justification or alternatives should be presented."

APPENDIX B

SUPPLEMENT 1 REFERENCES TO INTEGRATION OF INITIATIVES

APPENDIX B

This appendix contains some relevant quotations, in pertinent part, from Supplement 1 stressing integration of initiatives.

Cover Letter, Page 2 -

"..........It has become apparent, through discussions with owners' groups and individual licensees, that our previous schedules did not adequately consider the integration of these related activities.........."

"In addition, you are requested to submit with it a description of your plans for phased implementation and integration of the emergency response activities.........."

Enclosure, Page 4 -

"3. COORDINATION AND INTEGRATION OF INITIATIVES

3.1 The design of the Safety Parameter Display System (SPDS), design of instrument displays based on Regulatory Guide 1.97 guidance, control room design review, development of function oriented emergency operating procedures, and operating staff training should be integrated with respect to the overall enhancement of operator ability to comprehend plant conditions and cope with emergencies.........."

Enclosure, Page 5 -

"3.5 Specific implementation plans and reasonable, achievable schedules for improvements that will satisfy the requirements will be established by agreement between the NRC Project Manager and each individual licensee.........."

"3.8 The NRC recognizes that acceptable alternative methods of phasing and integrating emergency response activities may be developed. Each licensee needs flexibility in integrating these activities, taking into account the varying degree to which the licensee has implemented past requirements and guidance. An example of a way in which these activities could be integrated is discussed below.........."

"c. Using these EOP technical guidelines.....conduct a review of the control room design. Apply the results of this review to:.........."

"..........add additional instrumentation that may be necessary to implement Regulatory Guide 1.97.........."

APPENDIX C

EXCERPTS FROM OFFICIAL TRANSCRIPT OF 2/22/83 SUPPLEMENT 1 REGIONAL WORKSHOP

APPENDIX C

This appendix contains relevant quotations, in pertinent part, from the Official Transcript for the February 22, 1983 Regional Workshop on Supplement 1 to NUREG-0737 held in Arlington, Va.

Page 5, lines 12 to 16 -

"The basic set of requirements was laid out by the Commission. And it really is a basic set. There is a lot of guidance in the documents. A lot of backup NUREG guidance documents that are just guidance documents."

Page 9, lines 20 to 24 -

"In addition to a proposed schedule, the Commission asks that licensees and applicants submit a written plan of how they plan to phase the implementation of these various requirements and integrate them."

Page 69, lines 20 to 23 -

"Supplement 1 to NUREG-0737 requires that instrumentation be provided in the control room to assist the operators in preventing and mitigating the consequences of reactor accidents."

Page 75, lines 18 to 20 -

"The review efforts will treat only the exceptions to the Regulatory Guide identified by the licensee or applicant."

Page 86, lines 10 to 14 -

"We've been having 737 regional meetings now for two and a half years. I'm getting tired of them. The philosophical approach is to try to figure out what the requirements are, close them up and fix it and that's it."

Page 88, lines 14 to 25 and Page 89, lines 1 to 14 -

"MR. FADDEN: I'm Bill Fadden, Yankee Atomic Electric Company. You're just talking about Rev. 2. You mentioned the word requirements. Did you mean to use the word guidance?

MR. JOYCE: As you know all Regulatory Guides are guidances. We may have abused the word requirement. You're right. Regulatory Guides are guidances.

MR. FADDEN: The basic thing I guess we look at when we look at Regulatory Guide 1.97 and the basic standards that it endorses is whether you use it in procedures or not.

Would that be acceptable for selecting instrumentation rather than just a punch list that comes out of the back of a Regulatory Guide?

MR. JOYCE: Yes: as a matter of fact we probably encourage that, too.

MR. FADDEN: In other words, if we have the approved owners group procedures, if we show we do not use an instrument that's probably good justification for not including---

MR. JOYCE: Yes.

You could identify that underneath a common statement and give the full justification of why you believe in it and how to use a certain instrument based on your procedure, etc. and we'll review it at that time. I believe that it would be an acceptable basis."

APPENDIX D

REPRINT OF APPENDIX RECRIT FROM NSAC-1 SUPPLEMENT

APPENDIX RECRIT
ANALYSIS FOR POSSIBILITY OF RECRITICALITY

The Three Mile Island Unit-2 nuclear generating station is equipped with a variety of reactivity control features, designed for the purposes of keeping the plant within safe operating limits, under normal and abnormal service conditions. The reactivity control system design is tied to the station design basis, which includes a set of postulated transients or accident conditions. Since the Three Mile Island accident is believed to have exceeded the station design basis, questions have been raised as to the capability of reactivity control systems in maintaining the plant in a subcritical condition during the course of the event. Thus, the issue of recriticality has been addressed in the post-accident inquiry.

In this appendix the recriticality question is explored in terms of two rather broadly interpreted accident phases. The first phase extends from reactor trip through the initial core uncovery, but prior to significant core degradation or disarray. Nominally, this is the time period from 0400 to 0630. The second phase covers the balance of the accident period (i.e., after 0630). In this period, substantial reactor core disarray is believed to have occurred.

The subsequent discussion will conclude that there is little likelihood of recriticality or conditions approaching recriticality during the first phase of the accident. This conclusion is contrary to primary indications, construed by reactor operators, that the reactor may not have been adequately shut down (subcritical). For the second accident phase it is concluded that recriticality or near criticality was not likely to have occurred. However, the uncertainties in regards to both the dynamics and extent of core degradation makes this conclusion less definitive.

First Phase (0400-0630)

In a normal reactor trip, control rods are inserted to the bottom of the core, and the power level begins to decay in accordance with the 80-second period, consistent with the longest delayed neutron group half life. The reactor power falls below the power and intermediate ex-core instrument ranges, entering the source range. A typical source range power decay is shown in Figure 1. The power decay continues in accordance with the 80-second period until intercepted by the base count rate, defined by the source neutron production and subcritical multiplications. At Three Mile Island Unit-2 two neutron sources are important in determining the normal count rate curve: (1) installed Am-Be-Cm start-up sources, located at diametrically opposite locations at the core midplanes; (2) photo-neutrons (γ -n) generated by interaction of high-energy fission product gammas (primarily Kr^{88} and La^{140}) with deuterium (D_2O). During the early accident period, the photo-neutron source is the most important; the installed sources fix the ultimate core level count rate after photo-neutron sources die away.

The actual power decay time history at Three Mile Island Unit-2 was quite different from the nominal shutdown curve, as illustrated in Figure 2a. Instead of breaking from the 80-second period and continuing a downward trend, at a slow rate of decay, the source range recording began turning upwards at about the 30-minute mark. This upward trend continued until the reactor operator secured the reactor coolant pumps (at 100 minutes after trip), whereupon the count rate abruptly dropped to the base count rate level. Almost immediately thereafter, the count rate commenced a steep rise, reaching a peak that is nearly three decades above the normal. The intermediate range instrument recording (not shown) follows the source range recording where the two instrument ranges overlap.

In the time interval the source and intermediate range instruments were near their peak values (approx. 0630), some of the in-core self-powered neutron detectors began to behave erratically. High currents on some detectors were suggestive of substantial neutron fluxes in localized core regions.

The reactor operators initiated a manual (precautionary) scram at 0420 and checked rod bottom indicators to assure control rods were properly inserted. This may have been in response to abnormal ex-core neutron detector readings. As count rates continued to rise, the operators requested boron analysis at 0605 and 0630. The successive samples gave boron concentrations of 700 ppm and 400 ppm. These concentrations were low relative to the normal boration requirements at the existing stage in the fuel cycle, and tended to reinforce notions that the reactor may not have been adequately shut down as power boron concentrations a few hours earlier were 1030 ppm. Emergency boration was commenced by the operators prior to 0640.

The ex-core detector readings, in-core self-powered detector data, and boron analyses all point to a reactivity problem when these data are interpreted at face value. Nevertheless, careful analysis of instrument behavior, given a general understanding of what was going on in the core at the time, provides an alternative explanation.

In the minutes after the reactor trip, the primary system water inventory began to decrease as fluid was lost through the stuck-open electromatic relief valve. At saturation pressure, steam voids began to accumulate in the system. As two-phase mixture was pumped through the downcomer and core, three effects were manifest: (1) less water in the core decreased the intrinsic neutron source reading; (2) decreased fluid density in the downcomer permitted more neutrons to leak out to the ex-core detectors; (3) increased leakage from the core reduced neutron multiplication.

In order to reconcile the three somewhat competing effects neutron transport analyses have been performed to explain the source range detector behavior. In the first set of calculations one-dimensional (ANISN) transport analyses were used to determine detector count rates for homogeneous voiding of the core and downcomer regions. This model is appropriate for understanding the source range recording (Figure 2a) during the period of time the reactor coolant pumps were running (up to 0140 hours after reactor trip.) The results from these calculations are discussed immediately below. This discussion is followed by a presentation of two-dimensional neutron transport analyses, appropriate for the period immediately after reactor coolant pumps were secured (at 0140 hours).

The results of ANISN calculations for homogeneous voiding of the reactor core and downcomer are summarized in Table 1. A series of calculations were performed at varying void fractions. The homogeneous assumption and one-dimensional transport analyses are assumed to be valid on the basis of pump operation, acting to mix and distribute steam voids throughout the core and downcomer regions. Core average temperature was assumed to be 500° and soluable boron concentration at 1030 ppm for these calculations. A nominal core geometry was used.

Comparing the peak detector count rate in Figure 2a at 0140 hours, it may be observed that the average void fraction in the core/downcomer region was somewhere between 40-50% just prior to securing the reactor coolant pumps. This value is generally consistent with independent estimates of void fraction, based upon two-phase pump performance.

The one-dimensional analysis results confirm that the dominant influence on detector response is voiding the reactor vessel downcomer. This contributes to an increase in detector efficiency which more than out-weighs the effect in loss of source and water moderator. The net result was increasing counts seen by ex-core detectors, even while the reactor was becoming more subcritical. Consequently, it is fair to conclude that while homogeneous voiding prevailed (i.e., when reactor coolant pumps were running) the reactor was actually less reactive than immediately after shutdown. The upturn in the source range recording was the product of increased detector efficiency, due to the accumulation of steam voids in the downcomer.

The picture is more complicated after the pumps were stopped and phase separation occurred (after 0140 hours). As forced coolant flow ceased, falling liquid temporarily filled the downcomer. This resulted in an abrupt drop in the detector count rate (c.f. Figure 2a). As the core commenced to boil down, the downcomer water level dropped and more of the core came into view of the neutron detectors (Reference Figure CI-6 Appendix CI). As water was boiled out of the core the γ-n source began to diminish. In addition, increased neutron leakage from the core caused a reduction in neutron multiplication.

Evaluation of these competing effects under the non-homogeneous configuration necessitated multi-dimensional neutron transport analyses.

The multi-dimensional transport problem was analyzed using a DOT code R-θ /R-Z calculation under a 42-group Hansen and Roach cross section format. Core average temperature, soluable boron concentration, and geometry were the same as in the one-dimensional analysis. Results are shown in Figure 3. The curve and values for K_{eff} in the figure are based upon an axial void fraction profile which has been revised. Nevertheless the general trends are believed to be representative.

The transport analysis suggests that the "unshuttering" effect accompanying the drop in the downcomer water level dominates until the downcomer water level drops to about 6 feet. This is consistent with the one-dimensional results for a homgeneously voided downcomer (and core). When the core water level drops below a certain point the loss in γ-n source tends to assert itself. This causes the curve to bend over (reference Figure 3).

The change in reactivity during core boil-down is relatively modest until the water level almost reaches the bottom. K_{eff} drops from .937 to about .88 and holds fairly steady down to about two feet. This analysis is based upon an assumed boron concentration of 1030 ppm. Concentration by core boil-off may have somewhat reduced these K_{eff} values.

The shape of the curve in Figure 3 is fully consistent with the source range curve in Figure 2a, after 0142 minutes. The drop in downcomer water level leads to an increased detector efficiency, which produces increased count rates. The reactor remains subcritical, and is less reactive than when it was filled with coolant.

The two-dimensional neutron transport calculations permit conclusions to be drawn which are similar in nature to the one-dimensional results: voiding of the core and downcomer regions will produce source range detector responses that are entirely consistent with the recorded plant data. Recriticality was unlikely, given fairly reasonable assumptions about conditions that prevailed and K_{eff} values obtained.

The neutron transport analyses were used to characterize ex-core neutron detector behavior. However, these analyses do not explain the high currents observed on in-core self-powered neutron detectors.

The analysis of in-core self-powered neutron detector behavior during core boil-down and heat-up also suggests that detector currents were not a product of core recriticality. As explained in Appendix CI, the rhodium-Inconel detectors are subsceptible to a thermionic effect at abnormally high temperatures. Recent oven tests indicate that the detectors develop a small positive current (< 50 na) up to about 1000°F, whereupon the current abruptly changes polarity, reaching large negative values at high temperatures (> 2000°F). Positive and negative currents were observed at Three Mile Island; however, the small positive currents obtained from oven test is less than recorded currents at Three Mile Island*.

Although the large positive currents that were observed at Three Mile Island have not been fully confirmed by oven tests, it is reasonable to conjecture that temperature, as opposed to neutron flux, is the dominant factor influencing their behavior.

Satisfactory explanation of low boron concentrations, determined from samples at 0605 and 0630, has been a continuing problem. In some post-accident analyses these low concentrations have been ascribed to "flashing" in the letdown line or other inadvertent means of deriving "unrepresentative" boron concentrations. Sample analyses are believed to have been correct, since independent analyses by different persons yielded essentially consistent results, using the 0630 sample.

It now appears that low boron concentrations are the product of boron dilution in the A loop side, caused by distillation of borated water in the core and the accompanying condensation of boron free steam in the A loop steam generator, (boron volatility is low).

Prior to securing the reactor coolant pumps, plant operators commenced feeding the A loop once-through steam generator (OTSG) secondary side to re-establish level in the operating range. Feedwater spraying onto the OTSG tubes provided an efficient condensing medium for steam generated in the core; it is believed

* Oven tests were performed without the presence of gamma radiation, and it is believed that the radiation may accentuate positive currents at the elevated temperatures. Consideration is being given to experimental study of this behavior.

that a majority of liquid lost from the core during the initial boildown was transported into the A loop, rather than passing out the open relief valve. The net effect was a gradual reduction in boron concentration in the A loop on account of the dilution and increased boron concentration in the core. The imbalance in boron concentrations persisted at least until the reactor vessel had been refilled above level of the cold leg penetration.

Since chemistry samples are drawn from the low point in the A loop, it is not unreasonable to expect the low boron concentrations measured by the operators. Quantitative showing that there was no significant deficiency of boron in the core is underway.

A final argument relative to the boron concentration problem has to do with the effect on reactivity, given that such dilution of boron in the core actually occurred. According to the station safety analysis report, boron worth is figured at approximately 0.01% Δ K/K per ppm for an undamaged core. A reduction in boron from 1030 to 400 ppm should have increased reactivity by about 6%. However, rod worth inserted at reactor shutdown is in the neighborhood of 7%; transient xenon can be estimated at this time period at about 2%. On balance, then, the reactor would have been 3% subcritical after the supposed dilution (nominal core geometry assumed).

This assessment is approximate, and assumes an intact core geometry at a 500°F temperature. Other analyses(1) postulate different fuel damage conditions which give higher reactivity values. In some extreme cases (e.g., complete control rod and burnable poison rod destruction or removal) recriticality is possible.

Second Phase (after 0630)

After core disarray the recriticality question is difficult to answer conclusively, owing to uncertainties in fuel geometry. It has been superficially argued that recriticality is unlikely simply on the basis that any core degradation will represent a departure from a near-optimal geometry, designed for criticality in the first place. Conversely, it is possible to show (Nuclear Safety Guide TID - 7016) that 2.6% enriched uranium, optimally mixed with water moderator/reflector can produce a critical volume of under 70 liters (150 g/ℓ of UO_2); this is consistent with the station safety analysis report that a minimum of two clean moderated fuel assemblies are together sufficient to achieve criticality. Both extreme positions are likely to fall on either side of the range of conditions which actually occurred at TMI.

The case for or against recriticality must ultimately depend upon plant data analysis. Here, it is possible to show that recriticality is not likely to have occurred; however, it is not entirely clear whether or not core degradation may have substanitially reduced the margin of shutdown.

Nominally at least, an uncontrolled criticality would be accompanied by a sudden change in neutron count rates and (possible) evidence of energy release necessary to rearrange the fuel configuration into a subcritical configuration. Within the limits of resolution, the downward trend in the count rate should differ from the upwards trace, on account of the delayed neutron fraction.

Reviewing the source range instrument recording (Figure 2b) three candidate events are identified, occurring at 0747, 1350 and 1830. Among these the event at 0747 is the most interesting. That a significant energy release took place is evident by the overlay of other plant parameters, shown in Figure 4. A review of the sequence of events indicates that whatever happened at 0747 originated from within the core region and not from operator or equipment action outside. The event is likely to have occurred after core refill, since the high pressure injection system had been in operation for some 18 minutes prior to 0747.

The major difficulty in attributing the 0747 event to recriticality is the small variation in the source range signal; count rates only changed by about a factor of two. A simple thermal hydraulic analysis of the 0747 event suggests an energy release on the order of 2.3×10^6 BTU's. Assuming (conservatively) that fission produced this amount of energy over a 1 sec. time interval, power generation in excess of 80% full power would have been achieved. A spike in the source range, followed by decay in accordance with the 80-second period should have occurred; it did not. Moreover, pulses in the intermediate and power ranges should have been observed. None such were observed on the intermediate range. Power range data were recorded by the reactimeter at 3-second intervals; no statistically significant variations in power range detector current can be discerned. It is concluded that the 0747 event while yet unexplained, is unlikely to have been caused by recriticality.

The events at 1350 and 1830 are dismissed from consideration on the basis of: the small magnitude change in source range count rates and the lack of any significant energy release coincident with the event. Although evidence points against recriticality for these instances, it is worth pointing out that they were accompanied by small power range perturbations on the reactimeter. These perturbations are presently interpreted as being due to shielding variations caused by changing core water inventory, permitting fluctuations in gamma energy reaching the uncompensated detectors.

It may be concluded, simply on the basis of the available plant data, that recriticality was improbable. This is an important conclusion. However, it does not address the possiblility that there may have been a significant reduction in the margin that the reactor was shut down. This could have been the result of change in core configuration caused by damaging events in the accident sequence. The following discussion explores the question of whether or not a substantive change in the margin of reactor shutdown might have transpired.

Although recriticality is considered unlikely over the time interval of concern, there are legitimate questions which relate to the margin of shutdown. Comparing the source range recording against the base count rate, Figure 2b, it may be observed that the source range value is high. The high source range count rates persisted for some time and were confirmed with scale measurements by M. Shultz (TMI Industry Advisory Group) and R. Ball (B & W) on 4/19 and 4/25. Both readings were in the neighborhood of 25 cps.

A study of the source count rate decay curve was performed by H. Richings (USNRC).(2) To interpret the observed count rate, it is necessary to subtract off the fixed source neutron contribution which derives appreciably from the installed neutron sources. Since the count rate at TMI ultimately decayed to a constant 5 cps, this value can be used as the base count rate level.

Richings compared the actual decay curve with an expression built around a 12.8 day half life. The 12.8 day half life corresponds to the decay of Ba^{140}, which is the controlling factor in the La^{140}-D_2O photo-neutron production.

Richings' comparison over the time period 4/13 - 4/30 is shown in Figure 5. The close resemblance between the curve and count rate data strongly suggests that photo-neutron production from the La^{140} decay governed the long-term decay process.

* The comparison between the TMI time history and the nominal decay curve is based on reactor trip at full power of an Oconee nuclear unit, fitted to the TMI recording. A normal trip of TMI-2 at full power is not available.

The study of source range count rate decay does not account for the high count rate levels that were observed. The high count rate may be due to one or a combination of three possibilities: greater source strength; greater neutron multiplication; increased detector efficiency. The variation in source strength can be ruled out on account of the fixed relationship between core power history and Ba^{140} production, which is invariant to subsequent core degradation.

The neutron multiplication factor was originally pursued by M. Shultz.(3) Essentially, the analysis compares the nominal count rate to observed counts for the equivalent source term, and nominal K_{eff}. That is:

$$K_{eff2} = 1 - \left(\frac{CR_1}{CR_2} \right) (1 - K_{eff1})$$

After 22 days (time of Shultz's analysis) the photo-neutron source count rate (CR_1) is certainly less than 5 cps. Nominal K_{eff} is estimated at 0.71. Therefore for an observed count rate of 25 cps, the actual K_{eff} must be greater than 0.94. This is indicative of a major change in shutdown margin. However, it is not suggestive of imminent recriticality

Evidence contrary to the reactivity theory was also developed by Shultz. In the period between 4/13 and 4/17 the primary system was deborated from 3400 ppm to 3000 ppm. This deboration should have introduced reactivity net worth in the neighborhood of 4.0% to 5.3% Δ K/K. This is enough to have caused a significant variation in the count rate (enough in fact to achieve criticality if $K_{eff} \geq 0.95$). The fact that no variation in the source range count rate was observed suggests that the reactor was actually far subcritical.

The remaining possibility is that the source range detector efficiency was somehow changed. This line of reasoning postulates a significant release in Ba^{140} from the fuel into the coolant. The Ba^{140} is presumed soluble, decaying to soluble La^{140}. Some of the La^{140} finds its way into the downcomer annulus, producing photo-neutrons that are readily detected by source range instrumentation. The detector efficiency is increased in the sense that photo-neutrons have been physically moved (from the core) closer to the detector (e.g., the downcomer). On the debit side, however, is their incapability for neutron multiplication outside the core region.

H. Richings (NRC) has performed a scoping study of the downcomer $\gamma - n$ postulation, based upon a primary sample La^{140} activity (as of 4/11/79) of 150 mc/ml(2). He concludes that detector efficiency for neutrons produced in the appropriate downcomer region must be on the order of 1.42×10^{-2}. This is considered rather high for the situation at hand.

Richings' work has been independetly checked and a supplemental analysis has been performed to estimate photo-neutron production directly in the primary shield. It is not possible, using simplified analyses, to justify the high source range count rate. Although both analyses are based on primary sample Ba^{140} concentrations, there is no evident reason to expect these concentrations are not representative of downcomer Ba^{140} content.

The evidence at hand suggests that source neutrons emitted directly from the downcomer may be the cause of high source range count rates, rather than caused by a variation in shutdown margin. The analysis is not conclusive, and refined calculations may be warranted. One consideration which should be borne in mind, however, is the fact that source range count rates ultimately dropped to the neighborhood of 5 cps. This is consistent with the base count rate which would be sustained by the two installed (AM-Be-Cm) neutron sources. The low count rate value that was ultimately reached means the reactor was sufficiently subcritical to start with, or somehow evolved that way by gradual insertion of negative reactivity. This would have to be achieved at a 12.8 day half life, coincident with Ba^{140} decay -- an unlikely possibility.

REFERENCES

1. "TMI-2 Post Accident Criticality Analyses", GPU Services Technical Data Report #049, 31 August 1979.

2. H. Richings memo to K. Kriel (USNRC), "TMI-2 Source Range Detector Count Rate", 5/11/79

3. Industry Adivsory Group (IAG) memo, #IA-23, "Examine The High Counting Rate of BF^3 Neutron Detector", M. Shultz, 4/20/79.

TABLE I
CALCULATED K_{eff}, DETECTOR EFFICIENCY, SOURCE AND COUNT RATE FOR HOMOGENEOUS VOIDINIG

STATE	K_{eff}	E*	SOURCE (NEUTRONS/SEC)	COUNT R.
1030 ppm B Rods Crit	1.0	-	-	-
1030 ppm B Rods In	0.9368	1.2×10^{-10}	1.43×10^{11}	284
20% (Voids)	0.9097	4.9×10^{-8}	1.14×10^{11}	615
40% (Voids)	0.8582	2.3×10^{-9}	8.60×10^{10}	1430
60% (Voids)	0.7665	1.33×10^{-8}	5.74×10^{10}	3282
80% (Voids)	0.6146	1.31×10^{-7}	2.88×10^{10}	9791
100% (Voids)	0.4900	6.85×10^{-4}	2.05×10^{8}	2750

*Detector Efficiency is defined as the ratio of neutrons detected to the neutrons generated in the core.

APPENDIX RECRIT

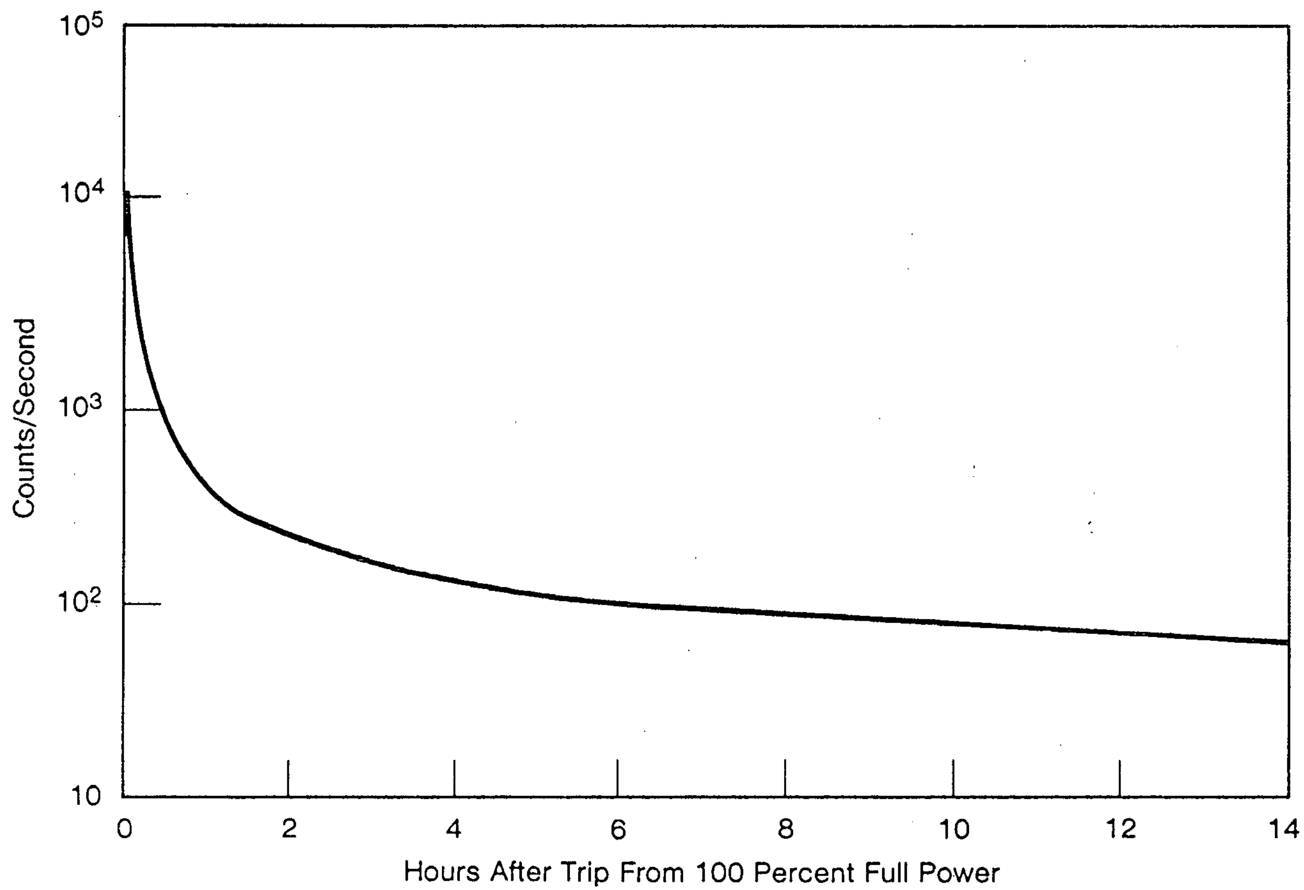

Figure 1. Typical Source Range Power Decay Curve

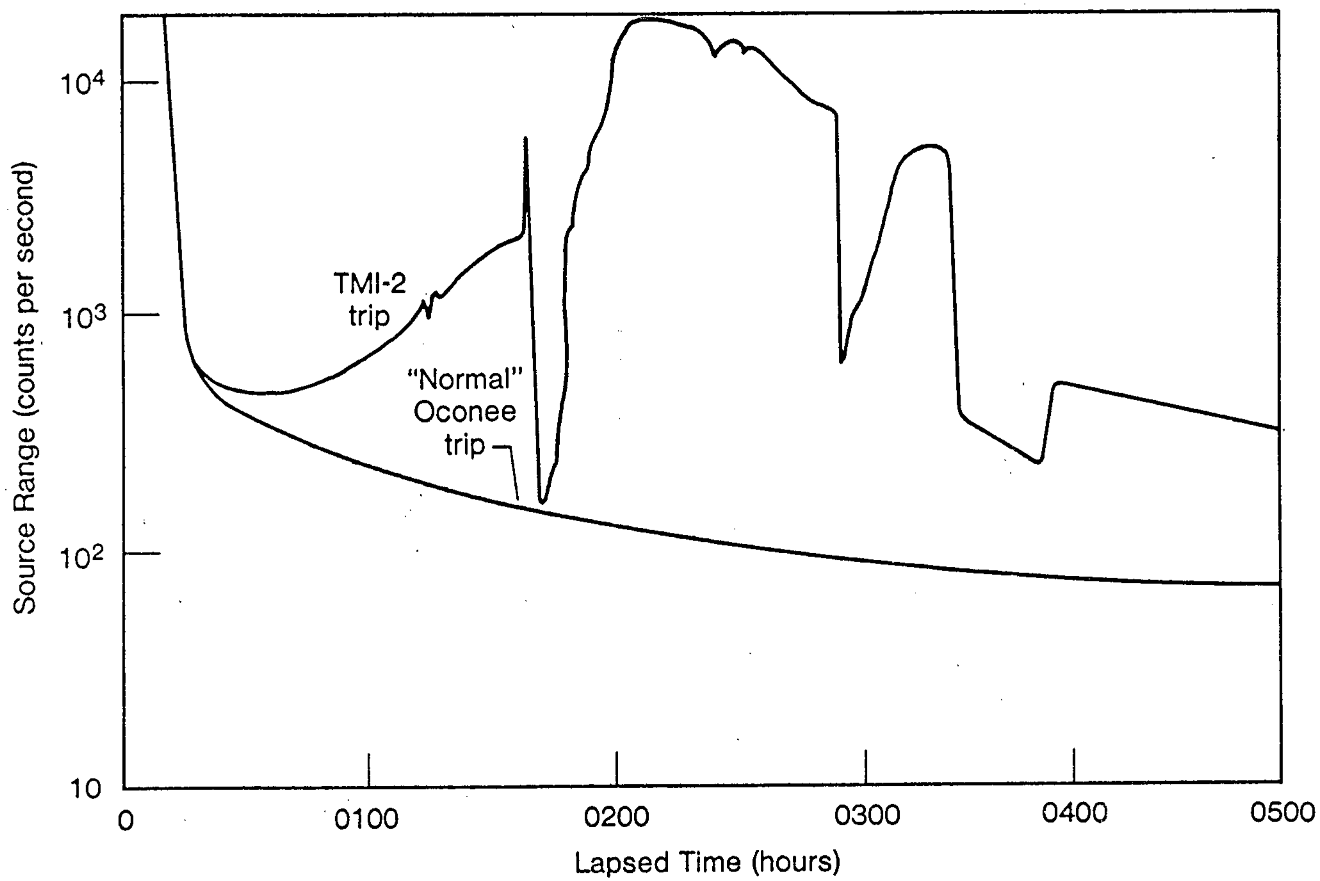

Figure 2a. Short Term Source Range Trace for TMI-2 (3/28/79)

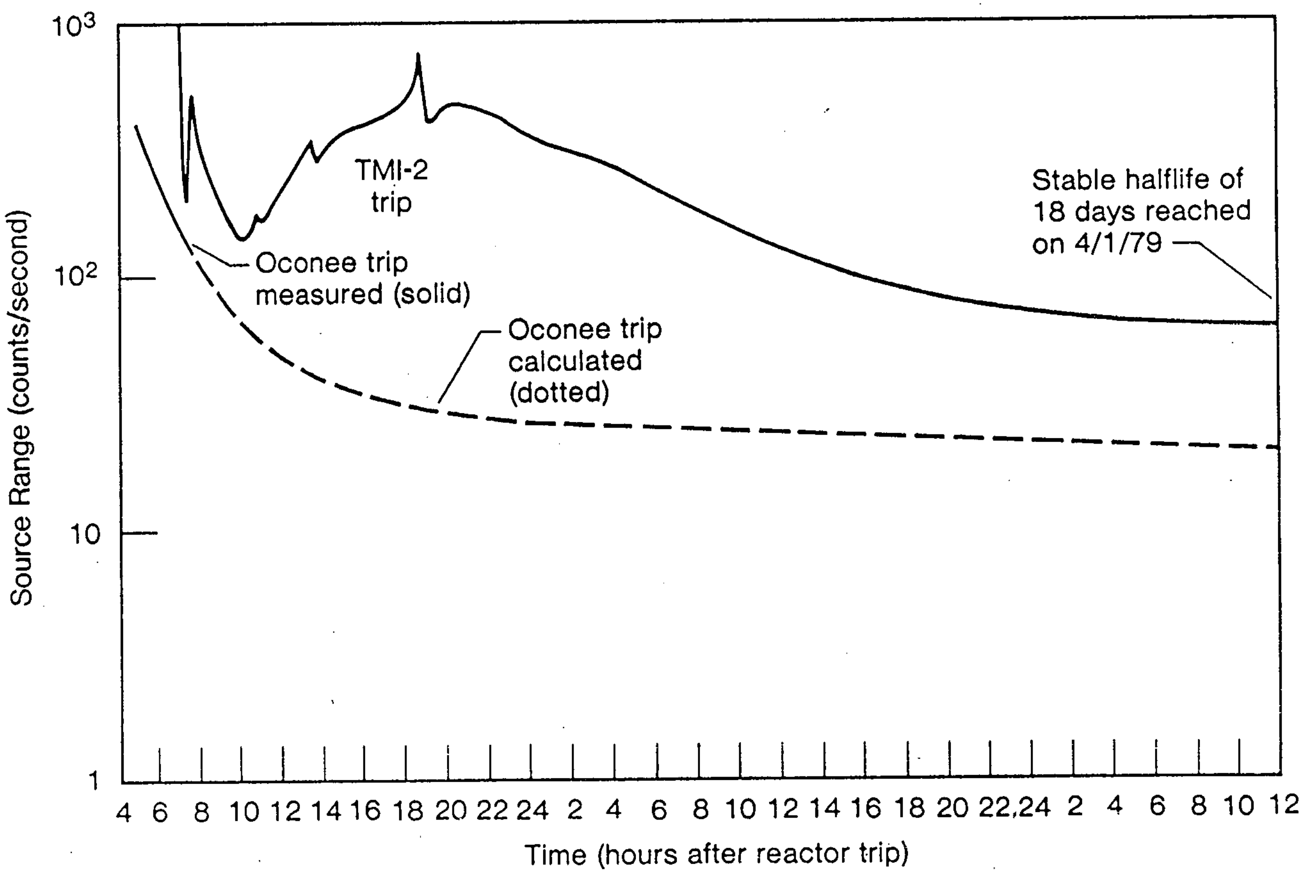

Figure 2b. Long Term Source Range Trace for TMI-2 (3/28/79–3/30/79)

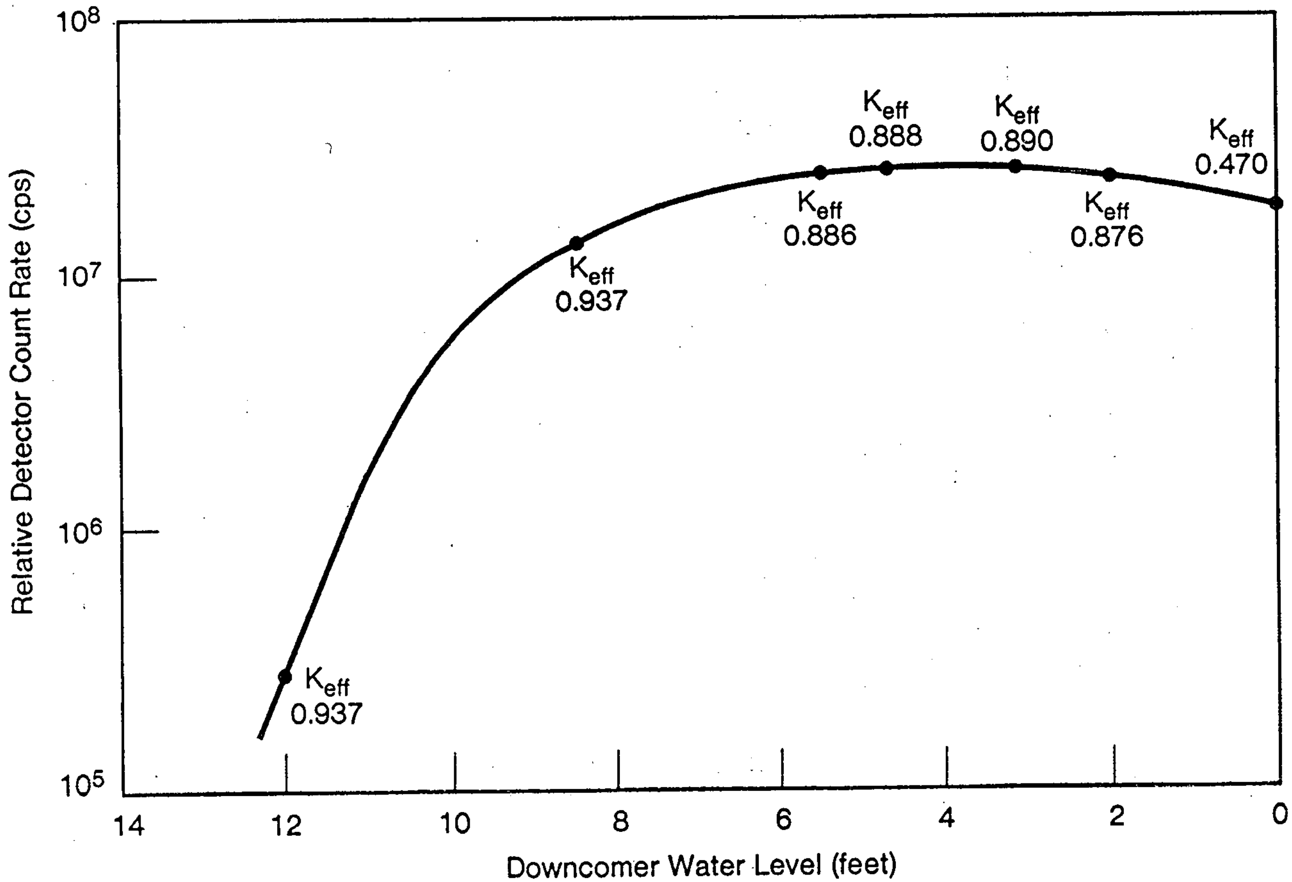

Figure 3. Downcomer Water Level Versus Detector Count Rate

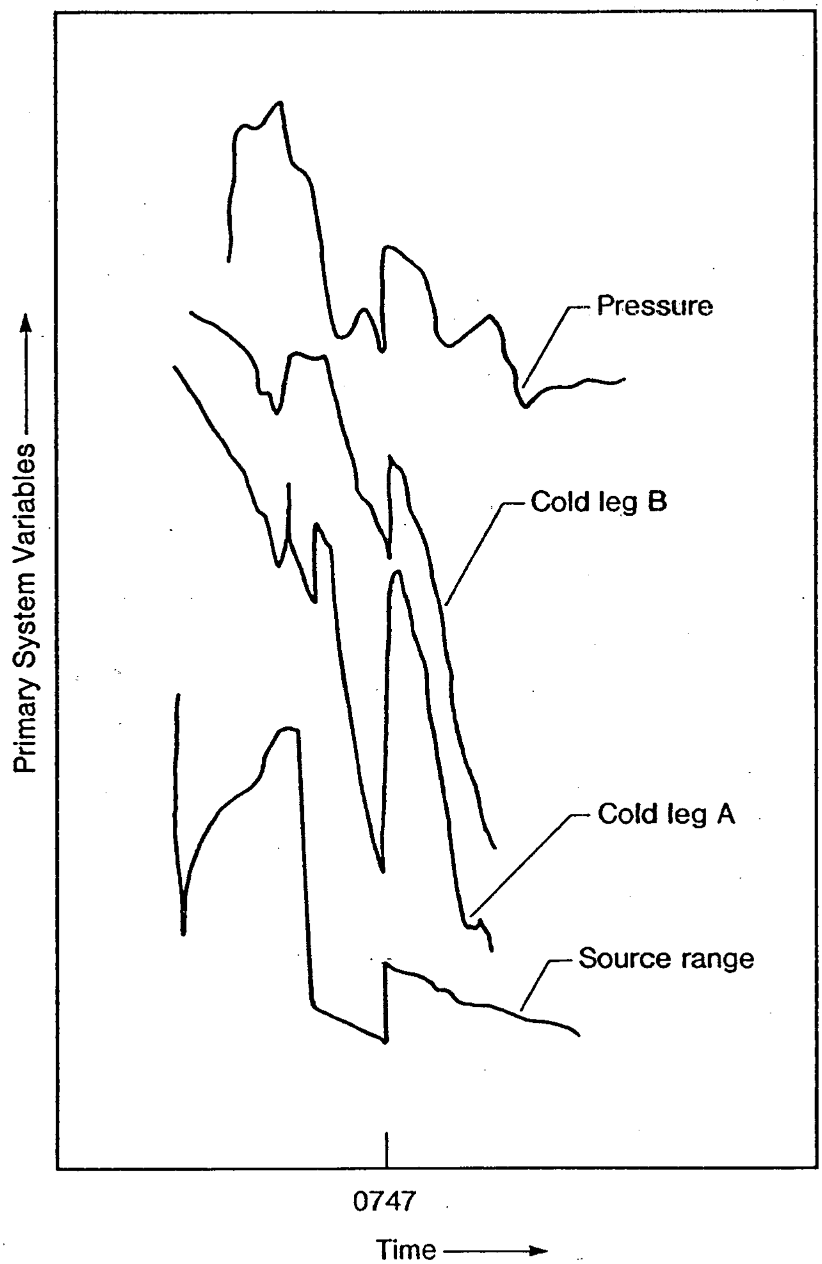

Figure 4. System Response to 0747 Event

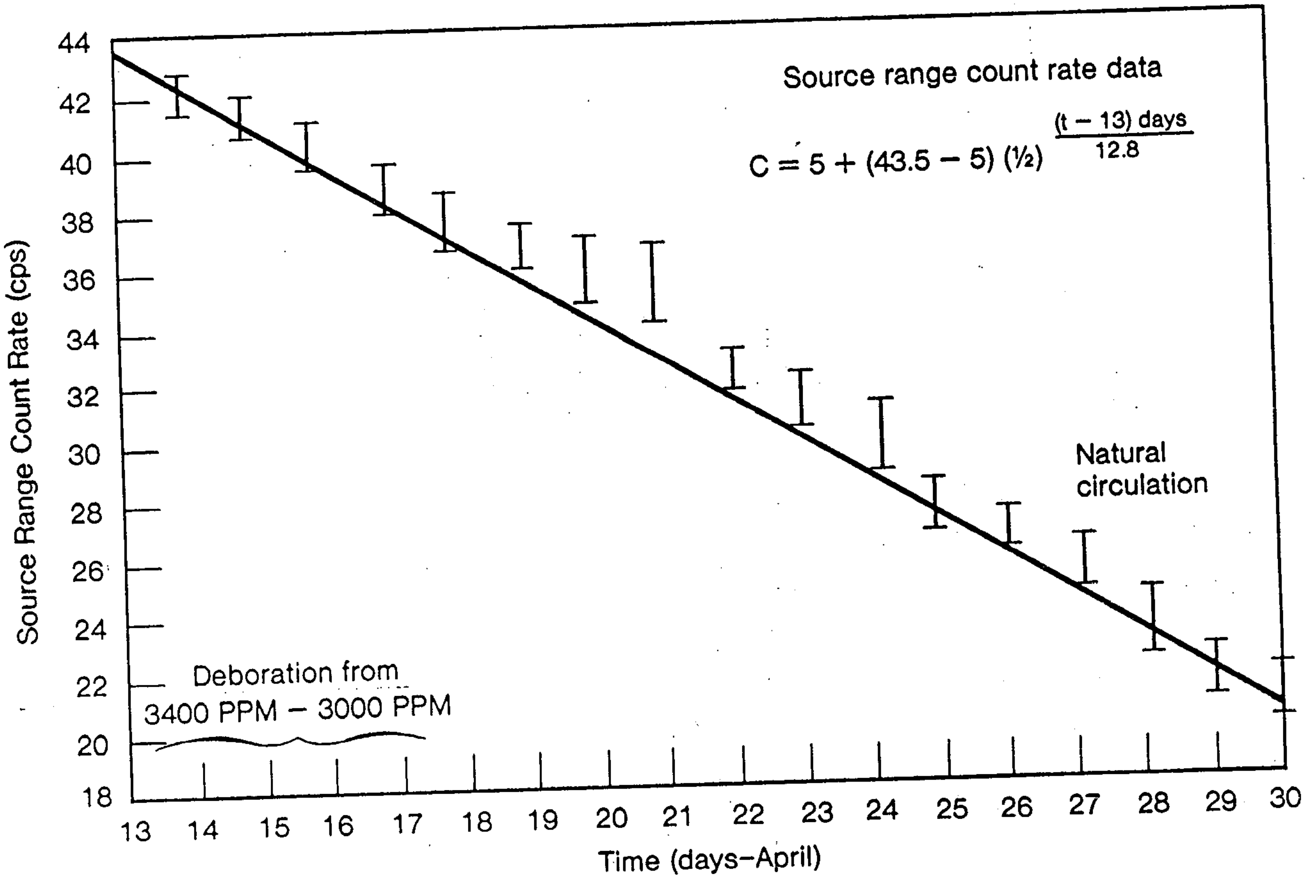

Figure 5. Comparison Against Source Range Count Rate Data

Reference: H. Richings (USNRC/core perf. br/DDS)

ATTACHMENT 2

STEAM GENERATOR WIDE RANGE
LEVEL INSTRUMENTATION

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.
INDIAN POINT UNIT NO. 2
DOCKET NO. 50-247
AUGUST, 1991

Background

The Indian Point Unit No. 2 (IP2) design includes four steam generators, two motor-driven auxiliary feedwater pumps (MDAFPs) and one turbine-driven auxiliary feedwater pump (TDAFP). The discharge piping is arranged so that each MDAFP supplies two of the four steam generators (SGs) while the TDAFP can supply all four SGs. Design basis accident analyses assume that only one MDAFP starts one minute after accident initiation and supplies 380 gpm for decay heat removal. There is one flow instrument per steam generator on the discharge side of the auxiliary feedwater pumps. Each SG is provided with one wide range and three narrow range (143 inch overlap with the top of the wide range) level instruments. The flow instruments and the narrow range level instruments are Type A, Category 1 and the wide range level instruments are Type D, Category 3. All three indications are recorded on the plant computer (Proteus) and SAS/SPDS.

Heat Sink Availability

Section 3, page 3 of the NRC Safety Evaluation states that "if the narrow range steam generator level were off scale low concurrent with an auxiliary feedwater pipe break, the operator would not be able to determine the status of steam generator heat sink availability." As explained above and in our September 12, 1986 submittal in regard to the auxiliary feedwater (AFW) flow instruments, the required heat sink is met with one MDAFP supplying two SGs. Therefore, two pipe breaks would be necessary to affect heat sink availability, which is highly unlikely.

Instrument Usage in Emergency Operating Procedures (EOPs)

The Westinghouse Owners Group (WOG) Emergency Response Guidelines (ERGs) establish the technical basis for the IP2 EOPs. Generic analysis performed in support of the ERGs demonstrates that adequate secondary side heat sink is maintained if

1) level in one steam generator is above the top of the tubes, i.e., in the narrow range, OR

2) if total feedwater flow to the SGs is greater than or equal to the capacity of one MDAFP.

In monitoring the status of the Critical Safety Function 'Heat Sink' and in the process of performing the EOPs, both AFW flow and SG narrow range level are used to verify adequate secondary side heat sink and each instrument is qualified to Category 1 requirements.

In the event of multiple failures beyond the design basis such that adequate secondary side heat sink is not verified, the operator is directed via the Critical Safety Function status tree F-0.3, 'Heat Sink', to FR-H.1, 'Response to Loss of Secondary Heat Sink', to determine if bleed and feed is required. Entry into FR-H.1 is not dependent on wide range level. Once

FR-H.1 is entered and normal containment conditions exist, bleed and feed is deferred until SG inventory decreases to a certain wide range level. If adverse containment conditions exist, bleed and feed cooling is initiated based upon narrow range level indication, without regard to SG wide range level.

This guidance is explained in the plant specific background document for FR-H.1 and is based on generic technical guidelines. The WOG recognized that qualified instruments were not available in all plants and provided direction in the ERG Background Document for FR-H.1 on alternate means to initiate bleed and feed cooling if the SG wide range level instruments are not qualified. If adverse conditions exist, narrow range level was determined to be the alternative symptom on which to base initiation of bleed and feed.

To summarize, wide range level indication is only used in a normal containment environment, where environmental qualification is not a concern, to initiate bleed and feed cooling of the core which is the last preferred method for heat removal. It is not the key variable for monitoring the operation of the SGs. The other verifications and required actions in the EOPs are based on narrow range level indication which is more conservative than using wide range level indication because action is taken sooner at a higher level.

Conclusion

Availability of the steam generators as heat sinks is verified by existing qualified instrumentation (narrow range level, AFW flow) and there are redundant heat sinks. At the February 22, 1983 Supplement 1 Regional Workshop, NRC staff indicated that an acceptable basis for not upgrading an instrument is to demonstrate and justify its use in emergency procedures. In accordance with this, since our procedures do not require its use in an adverse containment, it is not necessary to upgrade SG wide range level instrumentation from Category 3 to Category 1.

ATTACHMENT 3

CLARIFICATIONS TO SAFETY EVALUATION REGARDING
CONFORMANCE TO REGULATORY GUIDE 1.97, REVISION 2,
ISSUED SEPTEMBER 27, 1990

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.
INDIAN POINT UNIT NO. 2
DOCKET NO. 50-247
AUGUST, 1991

Clarifications to Safety Evaluation

Provided below are clarifications to the Technical Evaluation Report, "Conformance to Regulatory Guide 1.97: Indian Point 2", dated September 1989, which was attached to and reviewed in the NRC Staff Safety Evaluation Regarding Conformance to Regulatory Guide 1.97, Revision 2. These clarifications are discussed in the format of the Technical Evaluation Report (TER).

Paragraph 3.3.4 Degrees of Subcooling

Contrary to the statement on this item, review of this variable was not done as part of the staff's review of NUREG-0737, Item II.F.2, but was delayed at our request (July 3, 1984 submittal) by NRC letter dated February 6, 1985. By submittal dated October 18, 1988, we committed to upgrade the subcooling margin monitor to Category 1 requirements. Therefore, this exception is no longer required. However, the NUREG-0737, Item II.F.2 review still is required and closure of this item is herein requested. It may be appropriate to issue a Safety Evaluation for Item II.F.2.

Paragraph 3.3.7 Radiation Level in Circulating Primary Coolant

Note 13 in our original submittal (TER Reference 4) indicated that "this variable serves as a backup variable for monitoring fuel cladding breach" and the monitoring "methods include the delayed neutron gamma monitor and grab sampling (part of the post-accident sampling system) for analysis... Gross failed fuel detector indication is in the CCR." A later submittal (TER Reference 5) deleted reference to the gross failed fuel detector but neglected to delete the reference to the delayed neutron gamma monitor which is the gross failed fuel detector. This instrument was an experimental device and its use has been discontinued. The methods available, currently and at the time of our submittals, for monitoring this variable are an in-line isotopic analyzer and the grab sample, both of which are part of the post-accident sampling system. The isotopic analyzer, located in the waste gas compressor room in the primary auxiliary building, identifies and quantifies the radionuclides present in an undiluted sample, based on counts and energy levels. The gross activity is then determined from this information. This instrument is more accurate and more reliable than the gross failed fuel detector, therefore, it is concluded that the instrumentation supplied is more than adequate to monitor this variable.

Paragraph 3.3.10 Accumulator Tank Level and Pressure

The TER does not incorporate the information provided in a supplemental submittal (TER Reference 5), but since the Safety Evaluation indicated that this variable is the subject of a generic staff review, no further action is necessary at this time.

Paragraph 3.3.15 Main Steam Flow

Our original submittal indicated that both low and full range instrumentation were used for this variable but a supplemental submittal (TER Reference 5) deleted the low range instrumentation. The TER still refers to low and full range.

Section 2, "Review Requirements" states that "this report addresses only those exceptions to Regulatory Guide 1.97 that have been identified by the licensee." Section 3.3, "Exceptions to Regulatory Guide 1.97", states "the licensee identified deviations and exceptions to Regulatory Guide 1.97. These are discussed in the following paragraphs." As stated in the TER, our original submittal did not provide information on the main steam flow instrumentation but a supplemental submittal (TER Reference 5) did provide the information. Therefore, no exception was taken for this variable and there is no need to include it in the TER.

Paragraph 3.3.16 Auxiliary Feedwater Flow
Paragraph 3.3.17 Condensate Storage Tank Water Level

Section 2, "Review Requirements" states that "this report addresses only those exceptions to Regulatory Guide 1.97 that have been identified by the licensee." Section 3.3, "Exceptions to Regulatory Guide 1.97", states "the licensee identified deviations and exceptions to Regulatory Guide 1.97. These are discussed in the following paragraphs." As stated in the TER, our original submittal took exception from the Category 1 requirements but a supplemental submittal (TER Reference 5) indicated the instrumentation meets Category 1 requirements. Therefore, no exception was taken for these variables and there is no need to include them in the TER.

Paragraph 3.3.19 Containment Atmosphere Temperature

The TER reads "the licensee states that the emergency operating procedures do not utilize this variable." Our submittals (TER References 4 and 5) indicate that the variable is not relied on in the Emergency Operating Procedures.

The TER further reads "the licensee also states that containment heat removal is verified by other key variables, i.e., service water and component cooling water flow and inlet and outlet water temperatures, the temperature difference across the heat exchangers of these systems, and the residual heat removal heat exchanger outlet temperature." A supplemental submittal (TER Reference 5) deleted the instruments that measure component cooling water inlet and outlet water temperature, and, the temperature difference across the component cooling water and service water systems as a means of verifying containment heat removal. In addition, since Regulatory Guide 1.97 states "it is essential that key variables be qualified to the more stringent design and qualification criteria", we believe it is not appropriate to refer to the above instruments as key variables because they are only required to be Category 2 instruments, and are actually classified as such.

Lastly, the TER reads "the licensee states that the instrumentation will remain on-scale during all post-accident conditions that this instrumentation is designed for and expected to operate in." A review of our submittals could not locate this statement. Our September 12, 1986 submittal (TER Reference 5) does state that in regard to the containment atmosphere temperature instrumentation "the current range of $50^{o}F$ to $150^{o}F$ is sufficient to cover plant modes that are associated with Category 3 normal operating conditions."

Paragraph 3.3.23 Component Cooling Water Flow to Engineered Safety Feature System

The TER reads "the instrumentation will be on scale for any one or more component cooling water pumps in operation." Our submittals actually state that the "present range meets all operational requirements."

Paragraph 3.3.24 High Level Radioactive Liquid Tank Level

The TER reads "the licensee identifies a deviation in that the zero to 150 inch range encompasses all analyzed post-accident conditions..." The TER further reads, "based on the licensee's statement that the range is adequate to indicate the storage volume during all accident and post-accident conditions..." Our submittals actually state that the "range covers all anticipated operational occurrences."

Paragraph 3.3.28 Steam Generator Blowdown Radiation

Section 2, "Review Requirements" states that "this report addresses only those exceptions to Regulatory Guide 1.97 that have been identified by the licensee." Section 3.3, "Exceptions to Regulatory Guide 1.97", states "the licensee identified deviations and exceptions to Regulatory Guide 1.97. These are discussed in the following paragraphs." As stated in the TER, our original submittal identified this as a Type A variable but took exception from the Category 1 requirements, and a supplemental submittal (TER Reference 5) indicated the instrumentation is not required as a Type A variable. Therefore, no exception was taken for this variable and there is no need to include it in the TER.

Paragraph 3.3.29 Vent from Steam Generator Safety Relief Valves

The TER does not accurately reflect the information provided in our submittals. As stated in Note 35 of the August 30, 1985 (TER Reference 4) and September 12, 1986 (TER Reference 5) submittals, "the IP2 method for estimating releases through the atmospheric dump valves and the steam generator safety valves during the course of an accident is to obtain samples, taken upstream of the main steam isolation valves, of the entrapped noble gases for analysis in the on-site radiochem lab. Combining this information with total steam flow from the existing low range flow meters will yield the required data on quantity of radioactivity releases." In our October 26, 1988 submittal (TER Reference 6), we indicated that a backup for determining magnitude of release from these paths is to utilize "data obtained from the main steamline radiation monitors [and to combine] this information with total steam flow from the low range flow meters [to] yield the required data for calculating the magnitude and duration of release." It should also be noted that the main steamline radiation monitors and the low range main steam flow meters are not qualified to Category 2 requirements.

ATTACHMENT 4

UNREVIEWED EXCEPTIONS TO REGULATORY
GUIDE 1.97, REVISION 2

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.
INDIAN POINT UNIT NO. 2
DOCKET NO. 50-247
AUGUST, 1991

Unreviewed Exceptions to Regulatory Guide 1.97, Revision 2

Exceptions were taken for the following variables in our original submittal but the Technical Evaluation Report did not address them:

Analysis of Primary Coolant (Type C, Category 3)

A Canberra isotopic analyzer is located in the waste gas compressor room. Its readout is available in the radiochem/counting room. This is Category 3 instrumentation and 6.2(g) of NUREG-0737, Supplement 1 accepts displays in locations other than the CCR.

Pressurizer Level (Type D, Category 1)

Existing instrumentation range of 0 to 100% span covers 85% of total volume (tap-to-tap). Considering this large fraction and the severe non-linearity outside the tangent points, we have concluded that the present installation meets the intent of the Regulatory Guide.

Reactor Shield Building Annulus - Noble Gas (Type E, Category 2)

This feature is not in the design of IP2.

Condenser Air Ejector Flow (Type E, Category 2)

IP2 does not have this instrument. The design flow of 20 CFM will be applied to the release assessment.

All Other Identified Release Points - Noble Gas (Type E, Category 2)

There are no other unmonitored release points.