BEAVER VALLEY POWER STATION

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

Duquesne Light Company

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EXECUTIVE SUMMARY

Duquesne Light Company (DLC) 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 evaluation to provide the technical justification for the NRC to grant an exception to the recommendations of Regulatory Guide 1.97, Rev. 2. The safety assessment has shown that technical guidance can be provided to enable operators to directly monitor the critical safety function of subcriticality in an adverse containment by other qualified instrumentation without reliance on neutron flux measurements. Further, existing post-accident sampling and boration systems are capable of supporting this approach. Also, this evaluation has shown that under certain thermohydraulic core conditions, neutron flux readings can be misunderstood by an operator which is contrary to the objective of NUREG-0737, Supplement 1. Finally, an order of magnitude cost estimate shows that approximately $3.14 million would be spent 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, Rev. 2 and the requested exception should be granted.

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 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 DLC 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 equipment for Beaver Valley Power Station (BVPS) included a review of R.G. 1.97, Rev. 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. It was determined that it is not 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.

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

This report presents a regulatory and technical analysis of this issue and training guidelines for operators to use 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 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 continue to behave erroneously and 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 BVPS 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 to confirm that the core is subcritical that does not rely on neutron flux instrumentation 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 reading may not be indicative of core kinetics. 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 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. At the February 22, 1983 Regional Workshop, 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, Rev. 2 as a punch list.

The typical methodology for the Regulatory Guide 1.97, Rev. 2 review and NUREG-0737 integration used at BVPS included:

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

(2) A review of Regulatory Guide 1.97, Rev. 2 to develop a list itemizing types and categories of variables that are recommended;

(3) A review of BVPS 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.

DLC utilized Regulatory Guide 1.97, Rev. 2 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 Regulatory Guide 1.97. 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, Rev. 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, Rev. 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.

Regulatory Guide 1.97, Rev. 2 reviews were 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 recommended 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:

a. Instrument range, accuracy of measurement
b. Time interval during which the measurement is needed
c. Environmental Qualification
d. Seismic Qualification
e. Quality Assurance
f. Redundancy and sensor(s) location(s)
g. Power supply
h. Location of display(s)

The careful top-down integrated approach to post-accident monitoring instrumentation that was requested by the NRC and implemented by DLC concluded that the neutron flux monitoring instrumentation need not be upgraded to Category 1 and was acceptable with its current qualification status as Category 3. Further, it was concluded that neutron flux is not a Type A variable. Thus, DLC does not believe that it is necessary to upgrade the neutron flux monitoring instrumentation and has sought an exception. This report is intended to provide further supporting justification for the exception.

3.0 SAFETY ASSESSMENT

3.1 Design Basis Accident Analysis Evaluation

The accidents analyzed in the FSAR for BVPS 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 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 the unlikely criticality if it were to occur during the these events. Accordingly, technical guidance was developed to provide a systematic method to determine the status of the Subcriticality Critical Safety Function during adverse containment conditions. The basis for the technical guidance is fully described in Section 4.0 of this report.

During adverse containment conditions, the operators monitor 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. This guidance would be used as enhancements made to operator training.

A discussion of the accident analysis evaluation including the operator actions during EOP usage to determine subcriticality 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 Reactor Coolant System (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 into the containment 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 and create an adverse environment in which exposed equipment, 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, the equivalent of which is already in progress due to safety injection. The operator would then proceed, as directed by the EOPs, to monitor the Critical Safety Function variables and carry out other EOPs while boration is in progress. Since the LOCA scenario results in significant core heatup above 700°F, the operators would be directed to implement EOP FR-C.1 or C.2, determine that sump and RCS boron concentrations are above minimum shutdown values and 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-C.1 or C.2 and borate the RCS. 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 boron injection and recirculation is fully consistent with the LOCA mitigative strategy. 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 is sufficient action to assure that any approach to criticality is detected and that shutdown margin is maintained.

In the highly unlikely event that the reactor should return to power in the recovery phase following mitigation of a LOCA with a heat removal systems malfunction, EOP criteria will again direct the operator to function recovery procedures FR-C.1 or C.2, 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 is appropriately qualified to NUREG-0737.

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 manifested: (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 ex-core detectors; (3) increased leakage from the core reduces neutron multiplication. The second effect is by far the most dominant as far as ex-core detector response is concerned with 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. Additionally, it is important to recognize that voiding different regions of the core will have a varying effect on ex-core 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), references 5 and 6) 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 BVPS, reactor water level is measured by 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 which is presented in Appendix "RECRIT" of NSAC-1 Supplement issued in October 1979, which is reproduced here as Appendix C. 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.

Therefore, for the LOCA which yields an adverse containment environment, following the EOPs for determining subcriticality 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 is already installed at BVPS and 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 normal function for these events. The MSLB must occur inside containment in order for an adverse containment environment to be created. The technical guidance for determining subcriticality in an adverse containment will be used for this event.

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. The reactivity transient is mitigated by the automatic injection of borated water from the RWST. 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/Gs. Core exit temperature will be less than 700°F due to the rapid RCS cooldown and RCS temperature should then stabilize. 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 S/G level in the intact S/Gs and to control RCS temperature.

The automatic action of safety injection during this accident is to borate the RCS and accommodate RCS inventory shrinkage due to the rapid cooldown. 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 boration via the safety injection system and thus, no operator action is required to mitigate this expected initial reactivity transient.

For a MSLB 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.

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 to implement FR-C.1 or C.2 and initiate 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 BVPS FSAR.

By using core exit temperature and RCS wide range temperature indications, the status of the core is monitored directly and operator action to 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 Licensing Basis Accidents

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

1) Feedwater Enthalphy Decrease
2) Excessive Load Increase
3) Loss of Load
4) Loss of RCS Flow/Locked Rotor
5) Loss of Main Feedwater
6) Uncontrolled Rod Withdrawal
7) Startup of Inactive Reactor Coolant Loop
8) Rod Ejection
9) Steam Generator Tube Rupture
10) Inadvertent Boron Dilution

The evaluation was intended to define conditions that may need to be incorporated into the technical guidance discussed in Section 4.0 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 technical guidance. Additionally, 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 Westinghouse Owners Group (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 BVPS 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 BVPS EOPs go beyond the FSAR design/licensing basis accidents and transients upon which the plant design features are based.

Many of the EOP basis event scenarios involve adverse containment conditions requiring the use of new procedural enhancements, 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 hydraulic conditions. This review also included a determination 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 information was utilized in the table-top review to assess the changing hydraulic conditions during the event scenario.

The initial screening of 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.

As a result of the initial review, twenty nine (29) event scenarios beyond the standard design basis were identified 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. These event scenarios would cause the containment to become adverse with changing hydraulic conditions for which the EOPs would have to cope. The EOP usage was then evaluated in a table-top event by event review for each of the 29 beyond design basis events.

The 29 beyond design basis event scenarios 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 get into the scenario is largely the same 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 loss of subcooling occur during the scenario and continue at varying degrees throughout the event since the safety equipment built in BVPS 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 C 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 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 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 Assessment Results

A safety assessment of the design basis and beyond design basis accident analyses for BVPS was performed to determine it accident diagnosis and plant recovery can be successfully accomplished using the EOPs. For the worst case accidents such as loss of coolant (LOCA) and main steam line break (MSLB), an adverse containment environment would be generated and the operator would follow the EOPs for determining subcriticality. This does not rely on neutron flux instrumentation, but on direct reading core exit and RCS temperature instrumentation which are qualified. 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 EOPs for determining subcriticality in an adverse containment. Thus, successful diagnosis and accident mitigation can be achieved without the need to upgrade the neutron flux instrumentation.

3.4 Boration Requirements

The required boron concentration versus cycle burnup is calculated (and plotted) on a cycle specific basis for BVPS. 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 BVPS control room graphs book, are controlled by administrative and/or operating procedures, are referenced for use in the EOPs, and operators are familiar with their use having it 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.

3.5 Need for Upgrading Neutron Flux Instrumentation

Having determined that no safety benefit or increased protection will result from upgrading the neutron flux instrumentation in BVPS, even for events beyond the standard design basis event analysis defined paths, there is no cost beneficial reason to upgrade the instrumentation to Category 1. Additionally, since other non-adverse containment accidents can utilize the existing neutron flux instrumentation, there is no need for upgrading these instruments for those events. The cost analysis below shows that excessive expenditures would be encountered by DLC in order to make this upgrade.

3.5.1 Cost Analysis

3.5.1.1 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 DLC in response to adopting a regulatory retrofit item provide a functional response approach that was used in developing 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 3-3 and 3-4 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 3-2 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 seismic and environmentally qualified system to fully meet Category 1. 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 BVPS. Following the table is a brief description of the factors that went into determining the cost for each item identified in the table.

3.5.1.2 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.

3.5.1.3 Results Table Discussion:

Each bullet item in the Results Table is discussed below.

o 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)

o 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.

o 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 for installing a neutron flux monitoring system various categories were needed 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, 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 penetrations assembly upgrade.

o <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/or become 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.

3.5.1.4 Conclusion

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

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	Failure of more than 2 control System 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 (open) 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

TABLE 3-1

Functional Failures
(continued)

Symbol	Function	Definition of Function Failure
S/R-V^		Failure to open of all pressurizer safety and relief valves
S/R-VR		Failure to re-close of all pressurizer safety and relief valves
ECI	Emergency Coolant Injection	Failure to deliver borated water from at least 3 accumulators or any LHSI pump to RCS cold legs (large LOCA) or initial failure of both trains of HHSI
ECR	Emergency Coolant Recirculation	Failure to re-align to cold leg recirculation to inject water into RCS
ECI TERM	Emergency Coolant Injection Termination	Operator actions and/or equipment failures that terminate flow from HHSI pumps when its operation is still required
MSI	Main Steam Isolation	Failure to isolate main steam line to faulted (or ruptured) steam generator

TABLE 3-2

Beyond Design Basis Event Scenarios

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

TABLE 3-3

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 UFSAR
21. Modify structures in a radiation environment
22. Install, test and maintain hardware in a radiation environment
23. Draft license amendment

TABLE 3-4

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)

Typical Graph of Minimum Shutdown Boron Concentration Versus Cycle Burnup, At Cold, No Xenon Conditions

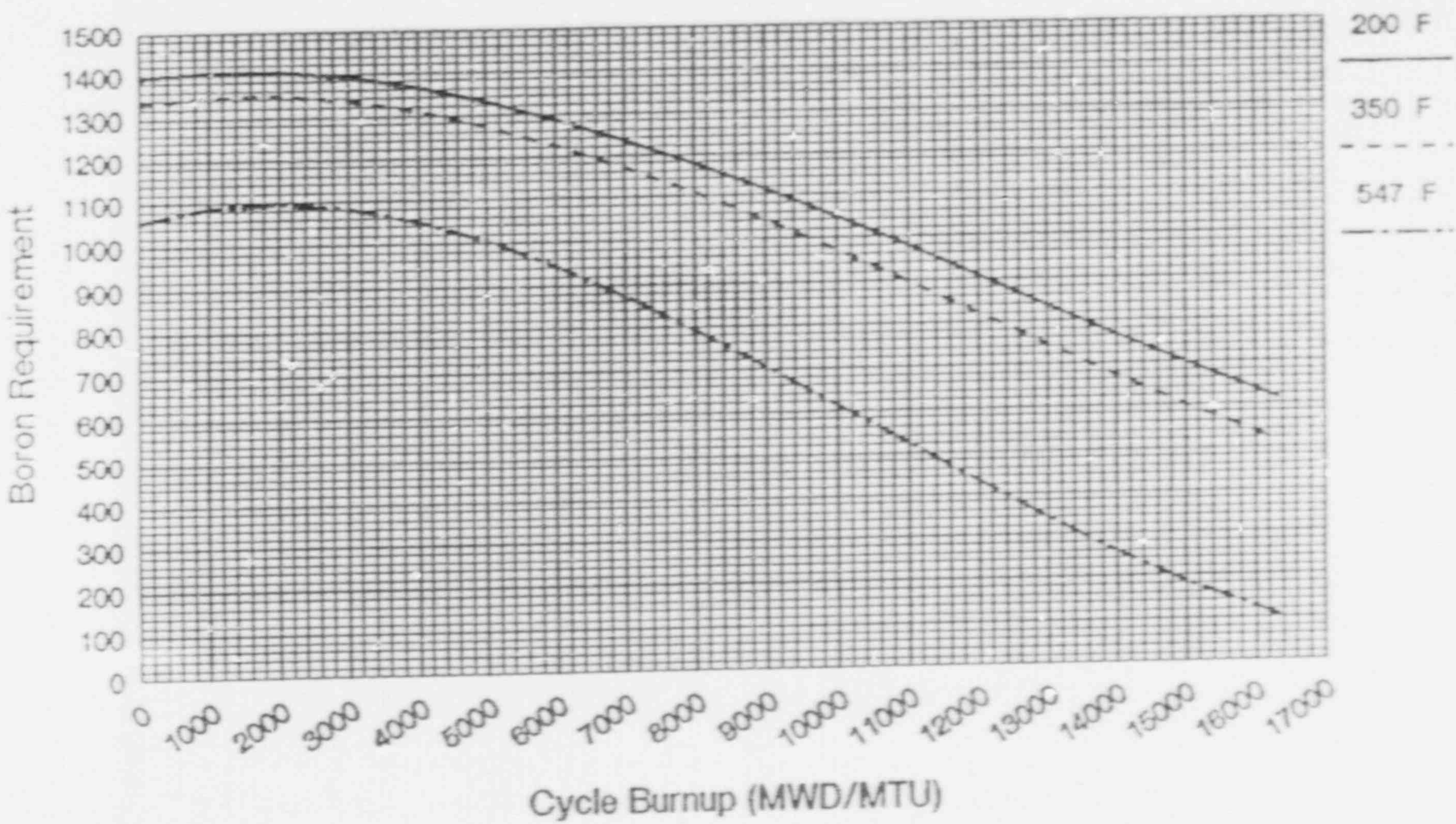

FIGURE 3-1

A NEUTRON FLUX MONITORING SYSTEM WITH AN APPENDIX R INTERFACE

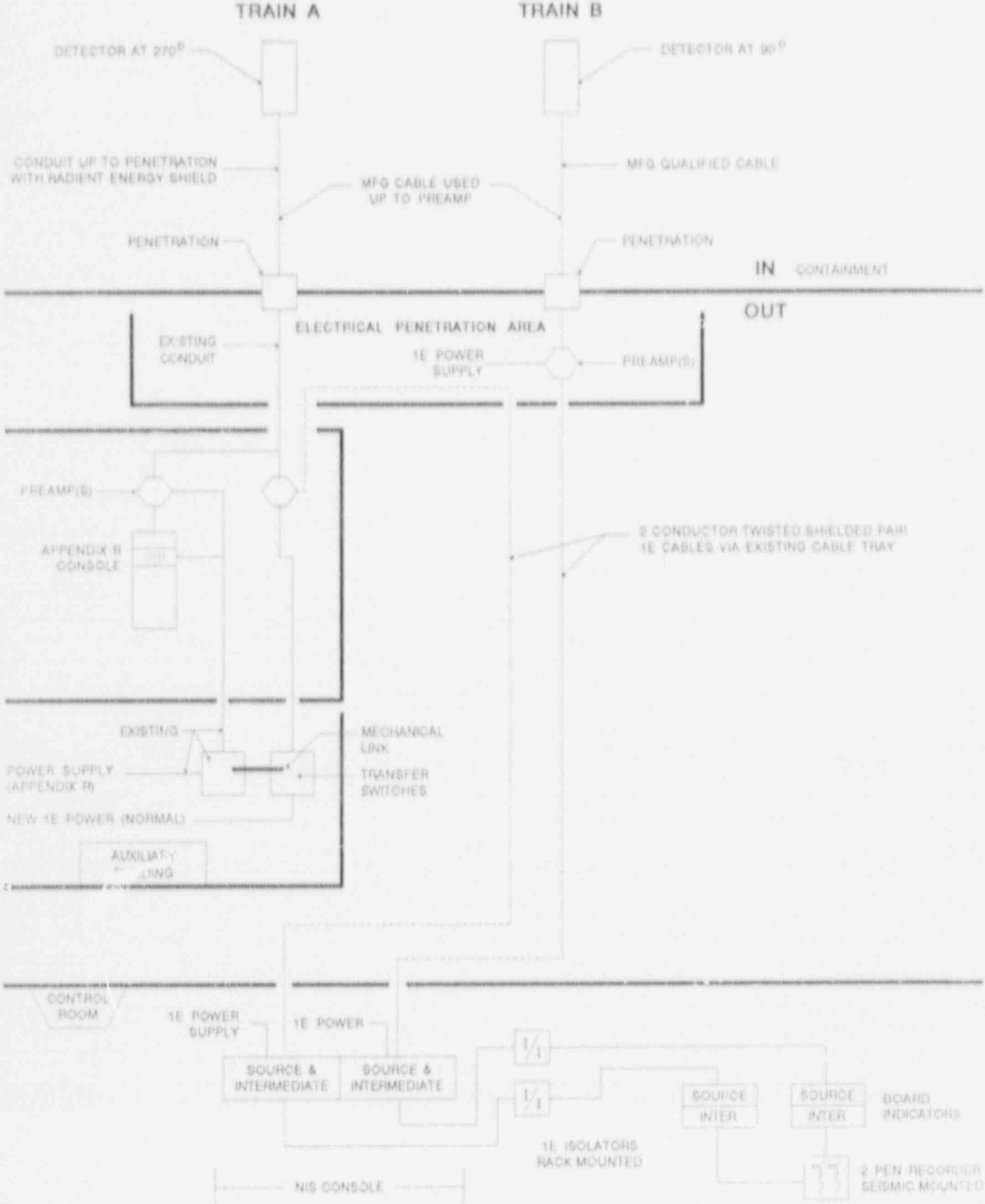

FIGURE 3-2

9GJD-101.GEM.063201NY

4.0 BASIS FOR TECHNICAL GUIDANCE

4.1 Criteria for Determining Subcriticality in an Adverse Containment

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 normal containment 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 trees represent 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, "Response to Nuclear Power Generation or ATWS" or "Response to Loss of Core Shutdown."

During adverse containment conditions, the operators 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 shutdown margin is verified by the operators and 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 as

identified in training are met. The basis of the technical guidance operator training is discussed below.

PROPOSED TRAINING ENHANCEMENTS

CRITERION: 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. Under a loss of reactor coolant condition, 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 boration should commence or continue.

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 approximately 700°F. However, because the Safety Injection Pumps, the predominant path for water to reach the RCS, will either be aligned to the RWST or the Containment Sump which are reliable sources for water with predetermined boron concentration, the boron dilution event is considered unlikely to have a significant effect on the rate of change of the boron concentration in the RCS.

Under a main steam line break condition, the coolant is rapidly cooled and core temperatures decrease sharply and significantly below 700°F until the blowing down steam generator 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 steam generator 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 boration should commence or continue.

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 Emergency Response Guidelines 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., boration) should be initiated unless the boron concentration is known to be above the minimum shutdown value. Once boration is initiated,

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 approximately 700°F. However, because the Safety Injection Pumps, the predominant path for water to reach the RCS, will either be aligned to the RWST or the Containment Sump which are reliable sources for water with predetermined boron concentration, the boron dilution event is considered unlikely to have a significant effect on the rate of change of the boron concentration in the RCS.

the operator can implement subsequent steps and carry out other procedures. Core exit temperature above 700°F in a core that is supposed to be shutdown, is considered a severe challenge to the fuel clad/matrix barrier. The operator would transition to the appropriate procedure for function restoration.

<u>INSTRUMENTATION:</u> Core Exit Thermocouples

PROPOSED TRAINING ENHANCEMENTS

CRITERION: 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: At this point, core exit temperature has been determined to be indicative of a potentially degraded core condition and a severe challenge exists. RCS pipe ruptures cause primary coolant to spill into containment and 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/C's (>700°F) is due to nuclear power generation in excess of the heat removal capability and the core must be shutdown. The operator would transition to the appropriate procedure for function restoration.

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.

INSTRUMENTATION

Boron analysis from the Post-Accident Sampling System. Control room graphs of minimum shutdown margin versus cycle burnup.

RCS Temperature.

PROPOSED TRAINING ENHANCEMENTS

CRITERION: 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: This criterion will be used to determine 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 additional reactivity insertion.

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. In the interim, until the boron concentration is confirmed, emergency boration may be necessary and the operator is directed to the appropriate procedure for a function restoration.

INSTRUMENTATION: Hot and Cold leg wide range temperature indication.

PROPOSED TRAINING ENHANCEMENTS

CRITERION: 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: Given that the plant accident may be at the point where core exit temperature is greater than approximately 700°F and the containment sump boron concentration is known and is greater than the minimum shutdown value, additional EOP confirmation of shutdown margin is established. 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, 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. The appropriate procedure for boration is followed in accordance with the EOPs.

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 the plant 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, 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 severe challenge to the Su[illegible]ction exists. The appropriate procedure for boration is follow[illegible] in accordance with the EOPs.

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, 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, 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 Temperature.

4.2 Emergency Operating Procedure Assessment

Revision 1A of the Westinghouse Owners Group Emergency Response Guidelines (Rev. 1A-ERGs) is the basis for the BVPS Emergency Operating Procedures (EOPs). 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 are acceptable as long as the differences from the generic guidelines are not safety significant. An assessment was conducted to verify that BVPS is within the technical basis of the generic guidelines of the Rev. 1A-ERGs and that the proposed training enhancements are fully consistent with the EOPs diagnosis and mitigative strategies and that no safety significant deviation exists. In fact, this review confirmed 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, an adverse containment training criteria that does not rely on neutron flux indication to determine subcriticality is appropriate.

The instrumentation used in an adverse containment condition includes:

a) core exit thermocouples;
b) 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 BVPS. Chemistry personnel are required to periodically demonstrate their familiarity with the Post-Accident Sampling System (PASS) equipment. The proposed training enhancements conservatively direct operator action based on direct reading of core

temperature and the knowledge of boron concentration. As such, the criteria are fully consistent with the ERG mitigative strategies and BVPS remains within the technical basis of the generic Rev. 1A-ERGs.

The generic guidelines and background documents were evaluated to determine the strategic safety considerations addressed by each guideline and the impact these have for determining subcriticality 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 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 establishes that boration be initiated in a conservative manner. Use of boron analysis by sampling is also used in many EOPs such as ES-0.1, ES-0.2, ES-1.2, ES-3.1, ES-3.2, ES-3.3, ECA-0.1, ECA-1.1, ECA-2.1, 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 training enhancements for determining subcriticality for adverse containment conditions that does not rely on neutron flux instrumentation for use in the BVPS operator training program.

5.0 POST-ACCIDENT SAMPLING SYSTEMS REVIEW

In accordance with the requirements of NUREG-0737, Item II.B.3, the Post-Accident Sampling Systems (PASS) installed at BVPS is designed to provide analysis of reactor coolant and the containment during normal and post-accident operating conditions. 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 samples points (e.g., hot legs, containment sump, RHR system, test points).

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:

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

The results of the PASS review concludes that the availability and reliability of the PASS and the required support systems is 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 BVPS.

6.0 EMERGENCY BORATION

6.1 Control Room Walkthrough

The emergency boration system was assessed at BVPS 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 facilitate the usage of controls by the operators. Human factors improvements such as better instrument scales and their placement, rearrangement of controls, clear labelling of alarms, etc. remain valid. Since no new emergency boration performance requirements are introduced by the proposed training enhancements and since the Control Room Design Review already included the emergency boration system, it is concluded that the Control Room operators can easily and effectively emergency borate the plant.

6.2 Availability and Reliability Assessment

The emergency boration system at BVPS 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 in a timely fashion to assure operability of the system and sub-systems, otherwise the plants are brought to a shutdown condition.

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

7.0 REACTOR VESSEL LIFE EXTENSION

Best estimate-LOCA technology allows a higher fuel linear heat generation rate (kw/ft) to the extent that outer periphery assemblies can be lower leakage than at present while permitting the plant to produce the same power level. The extremely low leakage core pattern results in less neutron embrittlement of the reactor vessel which thus extend its life. This work has been accomplished at several plants in a joint EPRI, Westinghouse, and utility program. Plants with low leakage patterns have experienced difficulty in properly measuring excore neutron flux. As a result, flux monitoring equipment had to be extensively re-calibrated and in some cases replaced for enhanced sensitivity.

Imposing the unnecessary requirement of Category 1 to be retrofit on the existing neutron flux instrumentation can impact the full benefits of best estimate-LOCA technology (extremely low leakage) to be realized, and certainly be a cause for further expenses later on to recalibrate and/or replace instrumentation in order to achieve required accuracy. These additional costs further support the justification that there is no cost-benefit (see Section 3.5) realized to upgrade the neutron flux instrumentation.

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23. Duquesne Light Company letter, J. D. Sieber to NRC, Document Control Desk, "Beaver Valley Power Station, Unit No. 1, Docket No. 50-334, License No. DPR-66, Generic Letter 82-33, Supplement 1 to NUREG-0737, Regulatory Guide 1.97, Revision 2, Response to Interim Review Results," April 22, 1987.

24. NRC letter, Peter S. Tam to J. D. Sieber, Duquesne Light Company, "Beaver Valley Unit 1 - Completion of Review, Inadequate Core Cooling Instrumentation (ICCI)," May 12, 1987.

25. Duquesne Light Company letter, J. D. Sieber to NRC, Document Control Desk, "Beaver Valley Power Station, Unit No. 1, Docket No. 50-334, License No. DPR-66, NUREG-0737, Item II.F.2, ICC Instrumentation System," March 18, 1987.

26. Duquesne Light Company letter, J. D. Sieber to NRC, Document Control Desk, "Beaver Valley Power Station, Unit No. 1, Docket No. 50-334, License No. DPR-66, NUREG-0737, Item II.F.2, ICC Instrumentation System," January 12, 1987.

27. Duquesne Light Company letter, J. J. Carey to Steven A. Varga, NRC, "Beaver Valley Power Station, Unit No. 1, Docket No. 50-334, License No. DPR-66, NUREG-0737, Item II.F.2, ICC Instrumentation System (Generic Letter 82-28)," April 24, 1984.

28. Duquesne Light Company, Operating Surveillance Test 1.49.2.

29. Beaver Valley Power Station Chemistry Manual BVPS-1/BVPS-2 CM, Chapter 6, Issue 1, Rev. 1 and Chapter 9.6, Issue 1, Rev. 0.

30. Duquesne Light Company letter, J. J. Carey to Director of Nuclear Reactor Regulation (NR), Item II.B.3; Post Accident Sampling System, August 31, 1982.

31. Beaver Valley Power Station, R.C.M. Chapter 5, Attachment - 1, REOP-2.1, Issue 2.

32. Beaver Valley Power Station, Drawing Nos. 8700-4M, 179A-16, 167A-12, 167B-14, 179B-7, 159A-17, 159B-10.

33. Beaver Valley Power Station Operating Manual, Chapter 14C, Issue 2, Rev. 6.

34. Duquesne Light Company letter, J. D. Sieber to NRC, Document Control Desk, "Beaver Valley Power Station, Unit No. 1, Docket No. 50-334, License No. DPR-66, Regulatory Guide 1.97, Revision 2, Response to Request for Additional Information," August 1, 1988.

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

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 instrurents 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 [illegible] downcomer.

The picture is more complicated after the [illegible] 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 possibility 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% $\Delta K/K$. 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.

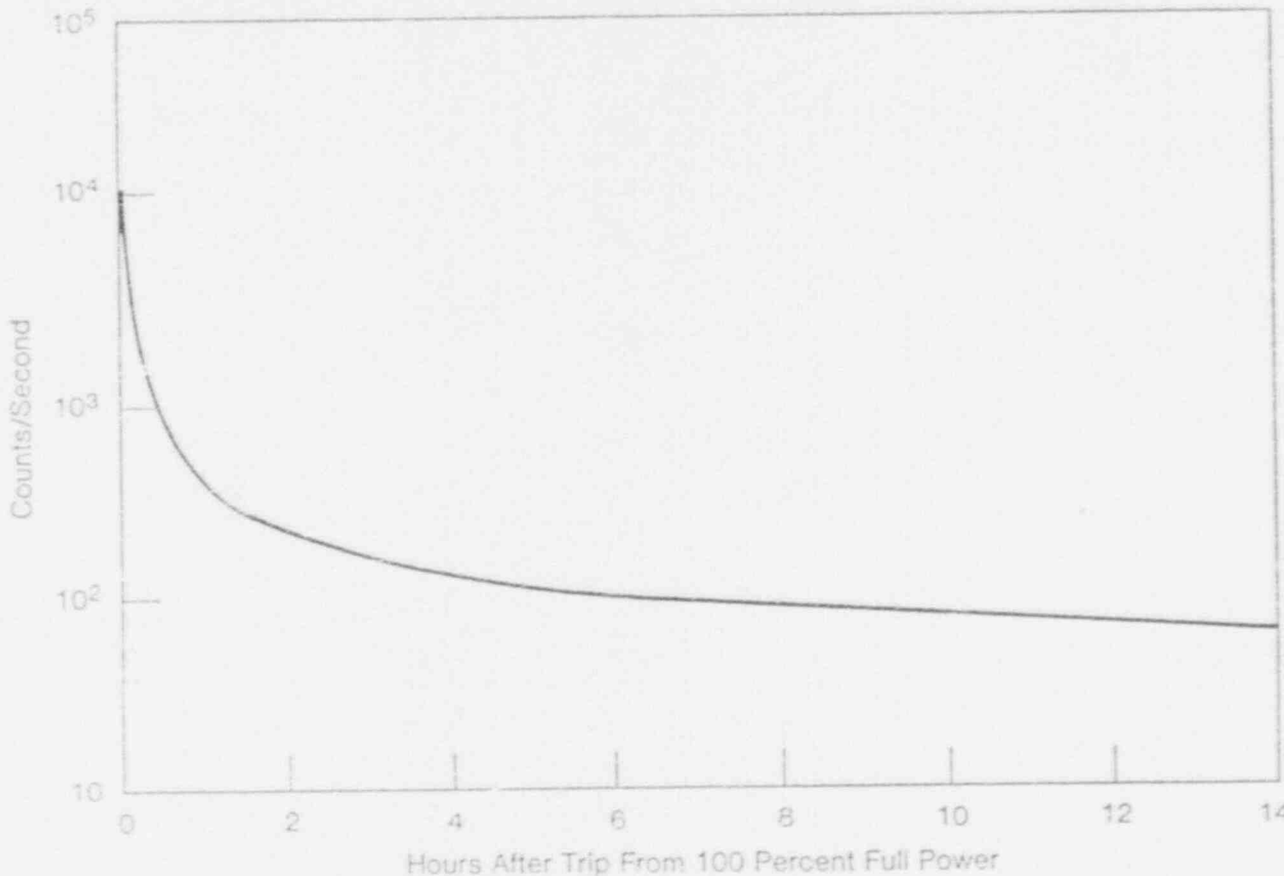

Figure 1. Typical Source Range Power Decay Curve

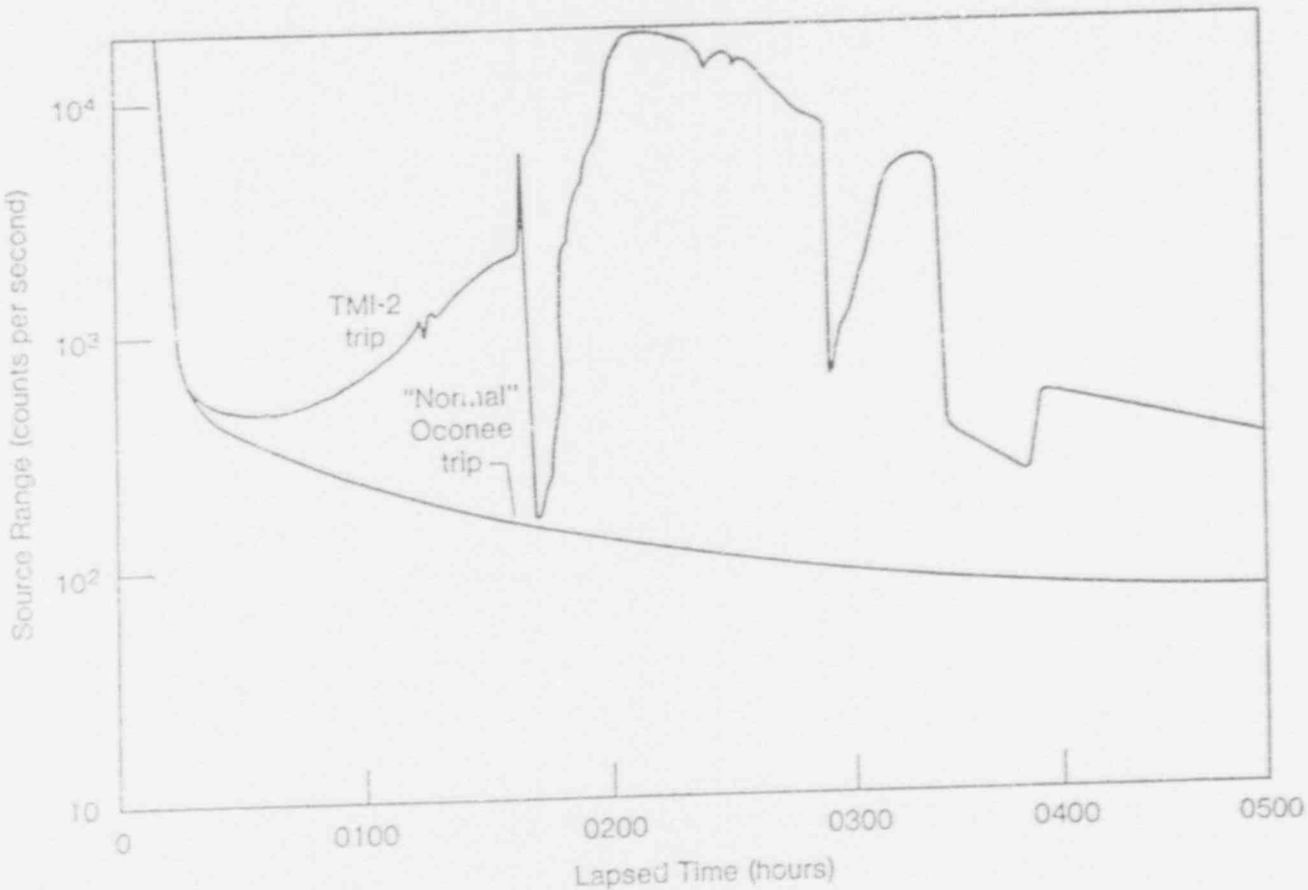

Figure 2a. Short Term Source Range Trace for TMI-2 (3/28/79)

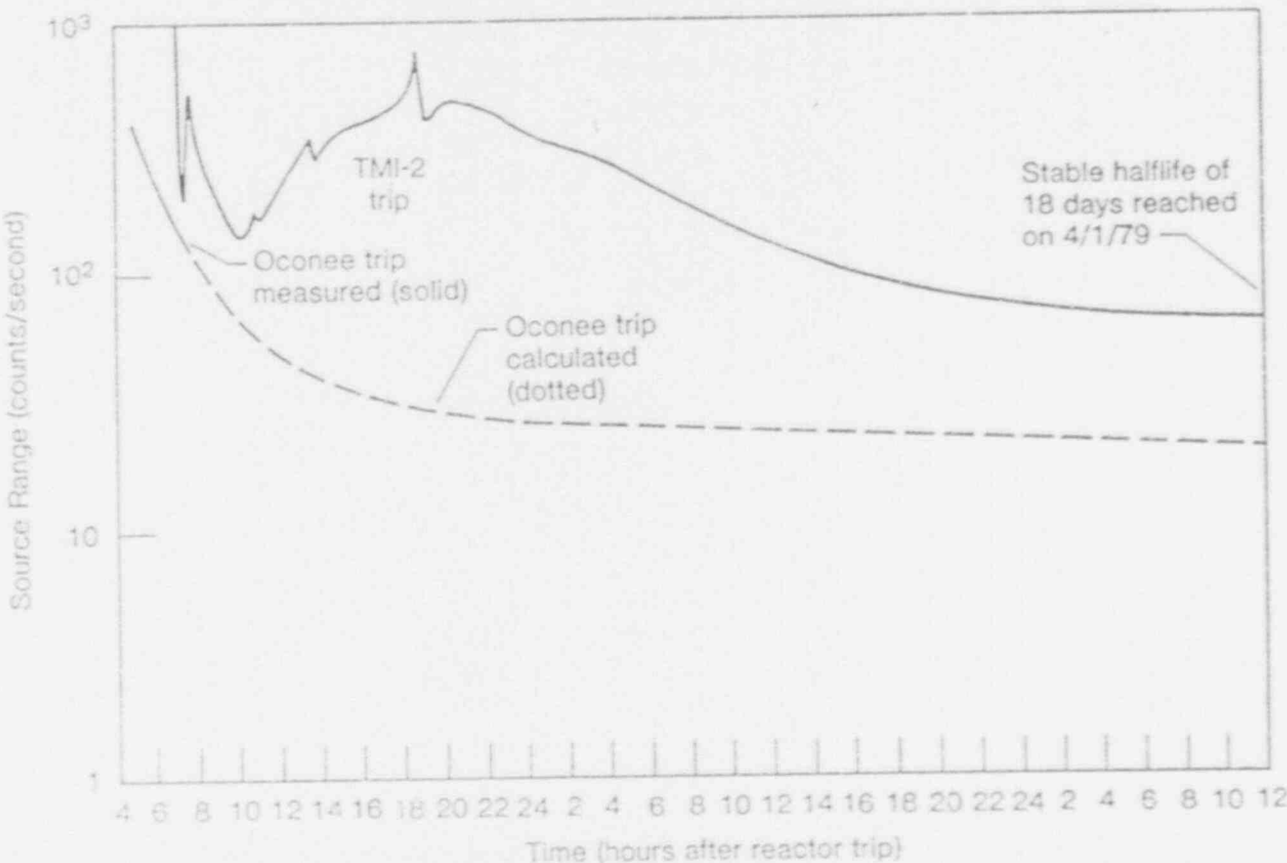

Figure 2b. Long Term Source Range Trace for TMI-2 (3/28/79–3/30/79)

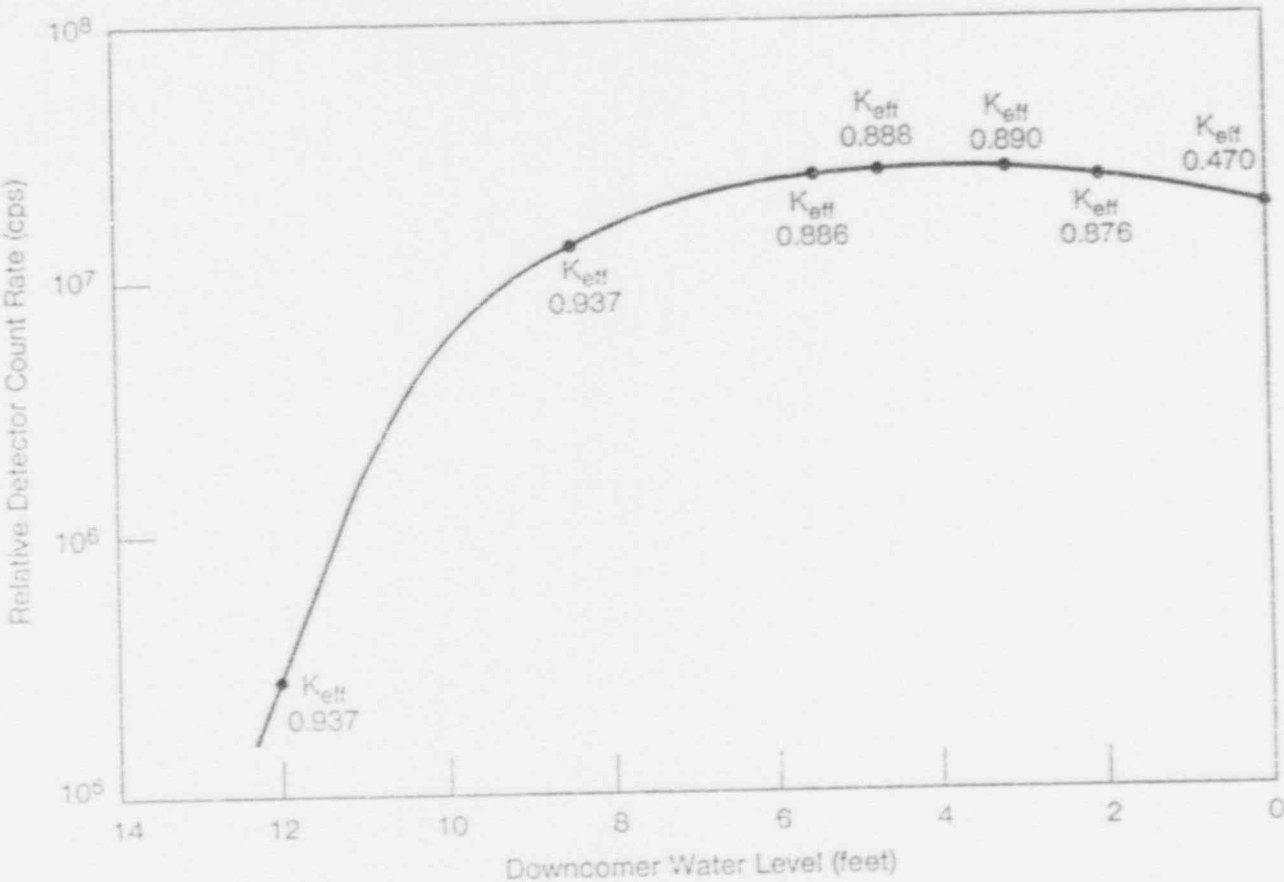

Figure 3. Downcomer Water Level Versus Detector Count Rate

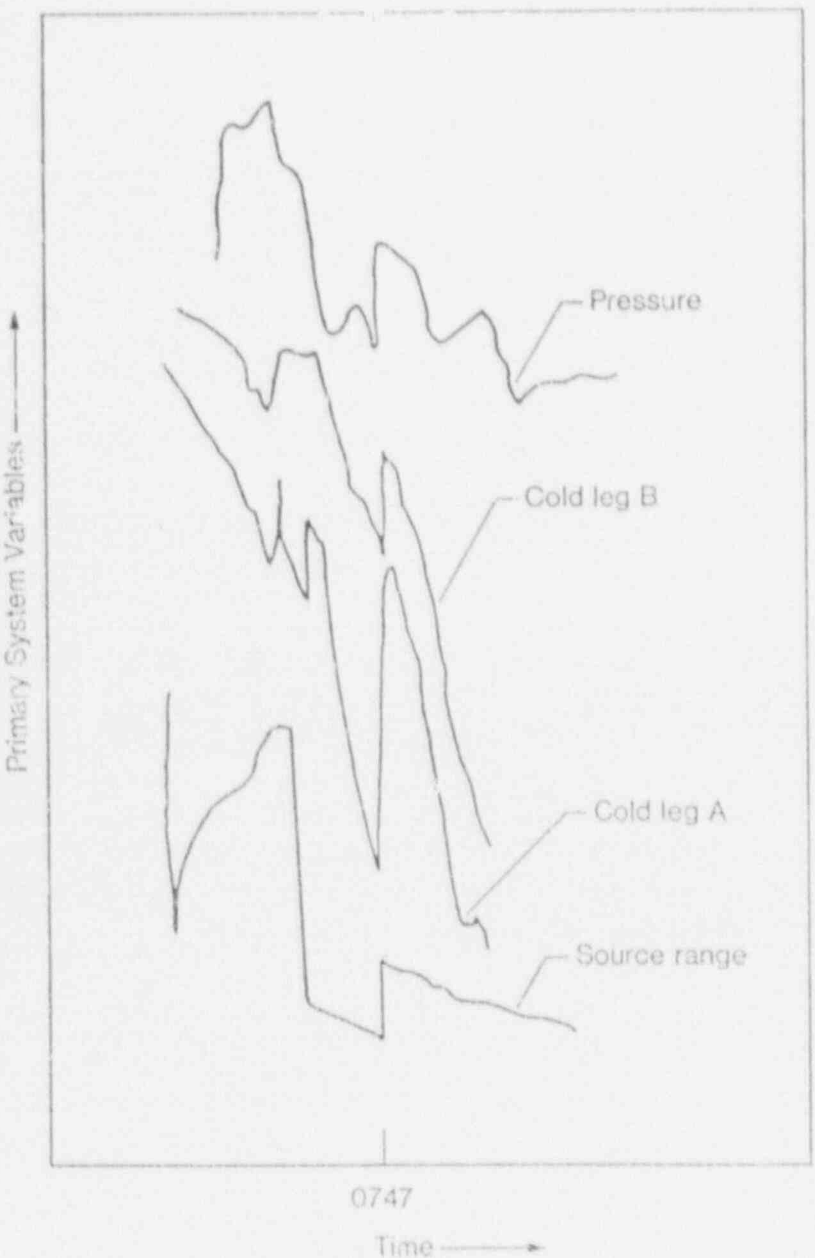

Figure 4. System Response to 0747 Event

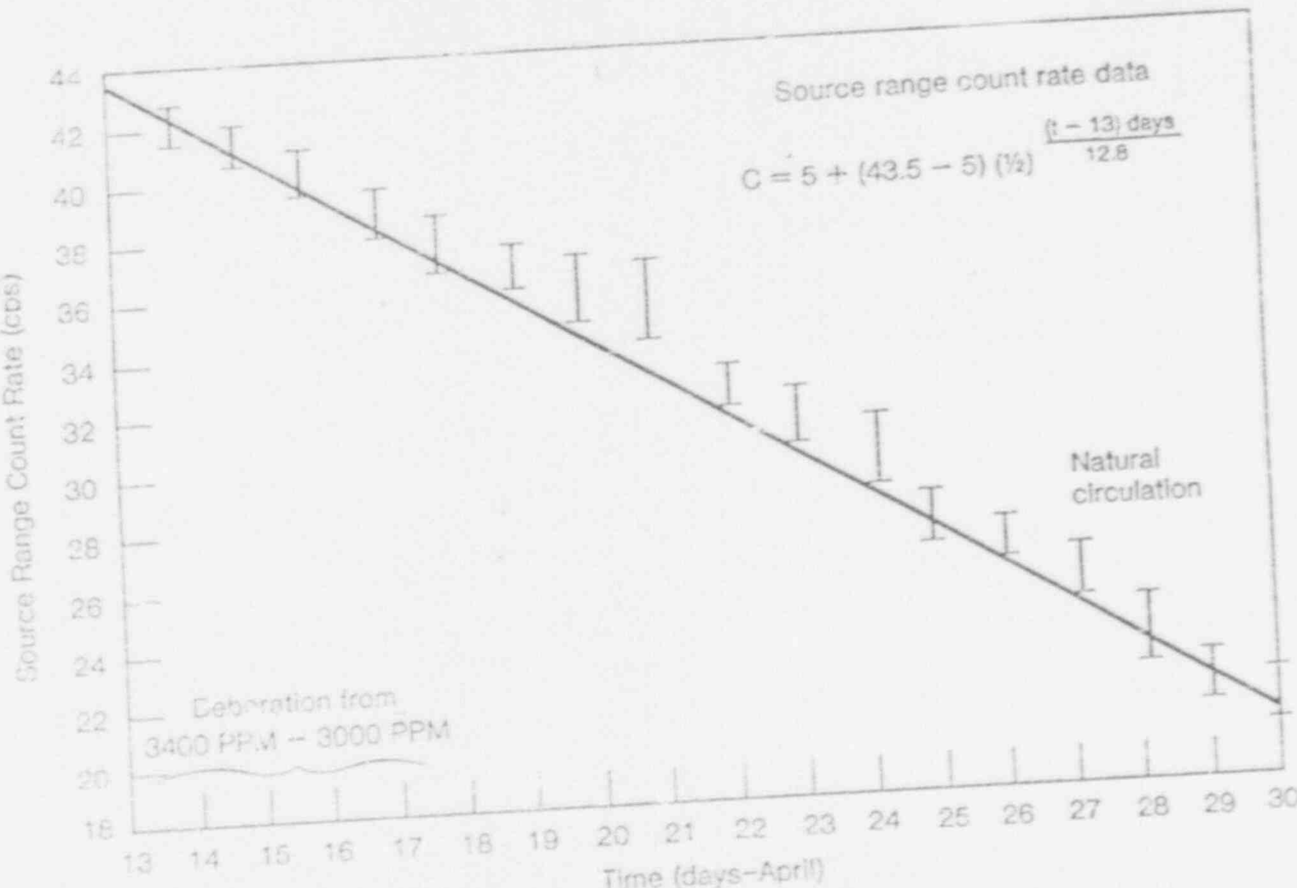

Figure 5. Comparison Against Source Range Count Rate Data

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