I. Introduction

This Appendix describes the compliance by the Shearon Harris Nuclear Power Plant with NUREG-0737, "Clarification of TMI Action Plan Requirements" issued by the NRC on October 31, 1980. Item numbers contained within this Appendix correspond to the item numbers of NUREG-0737. The information contained within this Appendix is in addition to and beyond the information contained in other sections of this FSAR. Where information between this Appendix and other sections of the FSAR is in conflict, this Appendix takes precedence.

This Appendix only addresses compliance with NUREG-0737 requirements. This, however, is only one portion of Carolina Power & Light Company's response to the accident at Three Mile Island Unit 2. Immediately following the accident, CP&L commissioned a Corporate Investigative Team (CIT) to review the design and operation of its operating plants and the SHNPP with regard to the accident and to make recommendations to management concerning modifications and improvements in design, operations, and training. In addition to the NRC mandated requirements, management has reviewed those recommendations made by CIT and they are being evaluated for incorporation into the SHNPP design, planned operations, and training. Additionally, CP&L has been very active in industry groups formed to respond to the accident including two NSSS owners' groups. The information obtained from these groups is continuously provided to those responsible for the design of the plant, the planned operations of the plant, and the training of plant personnel. Finally, as CP&L has implemented plant modifications and modified operations and training at its operating plants in response to the TMI accident, the experience and information obtained has been and is continuing to be provided to the SHNPP organization for incorporation into the design and operation of SHNPP. Carolina Power & Light's response to the accident at TMI is an ongoing process which will continue to implement lessons learned from the accident in the future as they are obtained.

II. Response to NUREG-0737, "Clarification of TMI Action Plan Requirements"

Carolina Power & Light Company's response to NUREG-0737, "Clarification of TMI Action Plan Requirements" is contained in the following paragraphs.

SHIFT TECHNICAL ADVISOR (I.A.1.1)

SHNPP complies with the requirements of this item. The Shift Engineer fulfills the requirements of the Shift Technical Advisor.

SHIFT SUPERVISOR RESPONSIBILITIES (I.A.1.2)

SHNPP complies with the requirements of this item. The Superintendent -Shift Operations fills the position described by NRC as Shift Supervisor. The required directions, descriptions, training programs, and procedures will be in place prior to receiving an Operating License.

SHIFT MANNING (I.A.1.3)

SHNPP will comply with the requirements of this item. The minimum shift crew composition in Section 6 of the Technical Specifications will be revised to be consistent with the interim guidelines. Administrative procedures establishing overtime restrictions consistent with the guidance of Generic Letter 82-02 will be implemented prior to receipt of an Operating License.

IMMEDIATE UPGRADING OF REACTOR OPERATOR AND SENIOR REACTOR OPERATOR TRAINING AND QUALIFICATIONS (ITEM I.A.2.1)
The requirement for an applicant for a senior reactor operator license to have been a licensed reactor operator for one year does not apply to cold license candidates. As noted in the NRR letter (3/28/80), pre-critical applicants will be required to meet unique qualifications designated to accommodate the fact that their facility has not yet been in operation. SHNPP will fully comply with the requirements of this item after cold licensing is complete.

ADMINISTRATION OF TRAINING PROGRAMS (ITEM I.A.2.3)

SHNPP complies with the requirements of this item (see FSAR Sections 1.8 and 13.2).

REVISE SCOPE AND CRITERIA FOR LICENSING EXAMINATIONS - SIMULATOR EXAMS (ITEM 3) (ITEM I.A.3.1)

SHNPP will comply with the requirements of this item by using the Harris simulator to give the simulator examinations. For cold license candidates the simulator operating examination will be given depending on installation and availability of the plant referenced simulator.

EVALUATION OF ORGANIZATION AND MANAGEMENT (I.B.1.2)

The Onsite Independent Safety Engineering Group (ISEG) functions are no longer performed by a dedicated group at the site. The functions of the dedicated ISEG are no longer required to be maintained, as these functions are redundant to established functions and processes performed by existing organizations. These functions are described in FSAR section 17.3.4.1.4.

SHORT-TERM ACCIDENT AND PROCEDURE REVIEW (I.C.1)

The Westinghouse Owners' Group, of which CP&L is a member, has developed and submitted to the NRC the required Emergency Response Procedure Guidelines. Following NRC approval of these guidelines, CP&L will make all necessary procedure revisions at SHNPP prior to fuel loading.

SHIFT AND RELIEF TURNOVER PROCEDURES (I.C.2)

SHNPP complies with the requirements of this item (see Section 13.5.1.3a(6)).

SHIFT SUPERVISOR RESPONSIBILITY (I.C.3)

SHNPP complies with the requirements of this item. The Superintendent -Shift Operations fills the position described by NRC as Shift Supervisor. The required directions, descriptions, training programs, and procedures will be in place prior to receiving an Operating License.

CONTROL ROOM ACCESS (I.C.4)

SHNPP complies with the requirement of this item (see Section 13.5.1.3a(7)).

FEEDBACK OF OPERATING EXPERIENCE (I.C.5)

An organization for the review and feedback of operational experience is in existence in the CP&L structure. Formal procedures for the dissemination and review of operational experience feedback information by the plant and corporate staffs are in place.

VERIFY CORRECT PERFORMANCE OF OPERATING ACTIVITIES (I.C.6)

SHNPP will be in compliance with the guidelines of I.C.6 before the issuance of an Operating License. Plant procedures will be in effect which specify the control of plant systems and the requirements to remove or return systems to service. Appropriate requirements for the use of a second qualified person will be included along with criteria for what constitutes a "qualified" person.

NSSS VENDOR REVIEW OF PROCEDURES (I.C.7)

SHNPP will comply with this item and commit to a review of low power and power ascension procedures related to the NSSS by Westinghouse personnel as discussed in NUREG-0737. These reviews will be performed by onsite Westinghouse personnel if they have the necessary expertise, otherwise, they will be reviewed by appropriate offsite Westinghouse personnel. CP&L considers this course of action to meet the intent of NUREG-0737.

The requirement for NSSS vendor review of Emergency Operating Procedures (EOP) is satisfied by Westinghouse's participation in the development of Westinghouse Owners' Group Emergency Response Guidelines (ERG). SHNPP EOP are based on Revision 1 of the ERG with only minor plant specific modifications. Due to the extensive Westinghouse review of the ERG, CP&L does not intend to submit the SHNPP EOP to Westinghouse for additional review.

PILOT MONITORING OF SELECTED EMERGENCY PROCEDURES FOR NTOLs (I.C.8)

CP&L will carefully consider the feedback developed from this NRC review of procedures and will incorporate any new requirements or other findings which appear appropriate to the SHNPP Emergency Procedures.

CONTROL-ROOM DESIGN REVIEWS (I.D.1)

A Control Room design review for SHNPP was conducted by the Essex Corporation from April 1980 to January 1981.

As a result of this review, the Main Control Board was redesigned and equipment relocated. Further, for design features which could not be evaluated or measured due to the then present stage of construction, Essex developed human engineering specifications. These specifications will be used as applicable to assure a well-engineered Control Room from the human factors engineering point of view.

CP&L considers this item complete.

PLANT SAFETY PARAMETER DISPLAY CONSOLE (I.D.2)

At SHNPP a top level display derived from plant parameters and indicative of the plant safety status will be displayed.

The display and data format (presentation) will be in conformance with human-factors criteria.

By using redundant instrument comparisons, checks for reasonability limits and normal operational limits, inferential checks to compare with other variables that are functionally related and by checking for detectable signal failures, all scanned variables will be quality tagged to indicate out-of-scan, open, out-of-sensor range, questionable status or substitute (entered) value.

By using reliable components, system design and redundant direct instrument channels on the main control board, essential indications will be available during normal and abnormal plant operating conditions.

Instrument channels, scan rates, processing methods and data displays will be chosen or designed to display pertinent information during both steady state and transient plant operation.

Signal processing (transforms), will be available along with sufficient display processing to allow all information to be displayed in the most suitable form. Instantaneous values, averages, summations, rates, and trends of any parameter or group of parameters may be chosen and displayed. Also, alarms and automatic display can be initiated based on instantaneous values or rates crossing setpoints or limits.

Interfaces between the SPDS system and safety systems will have isolation that meets all the required category IE requirements. The isolation will ensure the integrity of the safety systems. Further, where the SPDS interfaces with non-safety systems, the isolation and the SPDS design will be such that the SPDS is not degraded beyond the loss of signals from the failed non-safety system, i.e., the degradation will be limited to loss of information from the failed non-safety system.

Qualification programs will be included in the specification, design, implementation, test, installation and operation to demonstrate the SPDS's conformance to the design criteria.

Staffing

No one other than the normal control room operating staff will be required for operations of the SPDS.

Display Consideration

Instrument channels, scan rates, procession rates and display devices will be chosen to give information response equivalent to conventional dedicated control room instruments. The system will respond to transient and accident sequences.

Both the top level display and the supplemental displays will be designed to independently or jointly display the status of the plant. The displays will at all times be sufficient to display plant status to the control room staff.

Information will be displayed on a single primary display in a form that meets human-factors guidelines. The format(s) will be suitable for all plant operating modes.

Process signals available to the system (SPDS, TSC or EOF) will be available for display on secondary display(s) in a number of designated formats. The secondary information will be readily available to the RO, SROs, or STA to aid in the analysis of any event, to assist in terminating a possible event or to assist in mitigating the consequences of an actual event.

The basis for selecting the minimum data set for the primary display will be documented in the final system design.

The SPDS will access all available signals and will display information related to:

Reactivity control

Reactor core cooling and heat removal from primary steam

Reactor coolant system integrity

Radioactivity control

Containment integrity

Secondary displays will be available and system design will provide for automatic (event initiated) or manual (demand) control of the initiation or format of displays.

The system shall be designed to be modular in both hardware and software to allow for future modifications or additions. The system will be designed to be field expandable and testable to minimize any impact on system operation or integrity resulting from required modification or additions.

Design Criteria

Sensors, signal conditioners, and isolation devices shall be chosen from or designed to meet Class 1E requirements for all signals shared with or used by safety systems. The quality of all added devices (isolation, etc.) will be such that the quality of the signals to the SPDS will not be degraded.

Processing equipment, displays and related equipment will be high quality industrial grade equipment suitable for the application and the environment in which it will operate.

Controls, displays and related procedures will be designed to meet human-factors guidelines.

The system will be designed to function during and following an earthquake and still supply the essential data to the SPDS.

The function of the SPDS does not warrant seismic qualification because of the low probability of a seismic event concurrent with the need for the SPDS function, given the availability of seismically qualified displays for key safety parameters in the control room. Further, a separate additional concentrated display is not required as a backup for a nonseismic SPDS and is conceptually contrary to good human engineering practices.

Qualified indicators are available and with proper training of the operators, they are adequate for controlling the plant under all conditions.

The requirement to install separate additional seismic displays compounds the human factors problem and is also in conflict with sound operating practice which encourages that the operator use normal operating displays during accidents. This use of existing displays is most desirable since the operator will always get information to perform critical and normal operating functions from the same location. The SPDS, by definition is intended o concentrate a minimum set of plant parameters to aid the operator in the rapid detection of abnormal operating events. However, it is reasonable to use the normal qualified displays as a backup for this purpose.

TRAINING DURING LOW -POWER TESTING (I.G.1)

SHNPP is in compliance with this item. Our commitment to training during special low power testing is made in section 13.2.1.1.2H of the FSAR, Amendment 23.

REACTOR COOLANT SYSTEM VENTS (II.B.1)

For information on the RCS vent system refer to FSAR Section 5.4.12.5.

PLANT SHIELDING (II.B.2)

SHNPP compliance with this item is outlined in the FSAR Sections 12.2.1.12, 12.3.2.16, and 3.11.

POST ACCIDENT SAMPLING (II.B.3)

SHNPP compliance with this item is outlined in FSAR Section 6.2.5 and 9.3.2.

TRAINING FOR MITIGATING CORE DAMAGE (ITEM 11.B.4)

SHNPP will comply with the requirements of this item.

PERFORMANCE TESTING OF BWR AND PWR RELIEF AND SAFETY VALVES (II.D.1)

Carolina Power & Light Company is a participant in EPRI's PWR Safety and Relief Valve Testing Program. Test results have been submitted to the NRC by EPRI. Carolina Power & Light Company considers the program to be responsive to the requirements presented in NUREG-0578, Item 2.1.2, as clarified by NUREG-0660 and NUREG-0737, Item II.D.1.

Valves tested under the EPRI Program included actual valves (relief, safety, and block) from the SHNPP. Test conditions used were reflective of applicable operating/transient conditions of the SHNPP. Thus, these test results are directly applicable to SHNPP.

SHNPP will submit an analysis of the effect of as-built relief and safety valve discharge piping on valve operability by February 1984.

RELIEF AND SAFETY VALVE POSITION INDICATION (II.D.3)

The code safety relief valve positions will be directly derived from environmentally and seismically qualified reed type limit switches mounted on the valve top works and powered from a vital instrument bus. The position can be displayed on the Emergency Response Facility Information System (ERFIS) display located on the main control board. The ERFIS display will indicate the status as shut/not shut by graphic and/or English text displays. In addition, strap on resistance temperature detectors are provided upstream of the SRVs within the water section of each loop seal as a backup to indicate if there is valve seat leakage which is indicative of the valve not being fully shut. Both methods of detection are alarmed on the annunciator on the main control board.

The power operated relief valves utilize valve stem limit switches for position indication which is monitored by the use of indicative lights on their associated control modules. These limit switches are both seismically and environmentally qualified. The ERFIS display can indicate the status as open/not open by graphic or English text displays. In addition, a strap on resistance temperature detector is provided downstream in a common header as a back up to indicate valve seat leakage from any of the three PORV's. All methods of detection are alarmed on the annunciator on the control board.

<u>AUXILIARY FEEDWATER SYSTEM EVALUATION (II.E.1.1)</u>

The AFW System Safety Evaluation is addressed in FSAR Section 10.4.9.3.

<u>AUXILIARY FEEDWATER SYSTEM AUTOMATIC INITIATION AND FLOW INDICATION (II.E.1.2)</u>

SHNPP compliance with this item is described in FSAR Sections 7.2.2, 7.3.1, 7.3.2, and 7.5. Compliance with IEEE-279-1971 is detailed in FSAR Section 7.3.2.2.

<u>EMERGENCY POWER FOR PRESSURIZER HEATERS (II.E.3.1)</u>

SHNPP compliance with this item is outlined in FSAR Section 8.3.1.2.35.

<u>DEDICATED HYDROGEN PENETRATIONS (II.E.4.1)</u>

This item does not apply to SHNPP since qualified redundant in-containment thermal recombiners are provided. Refer to FSAR Section 6.2.5.

<u>CONTAINMENT ISOLATION DEPENDABILITY (II.E.4.2)</u>

SHNPP compliance with this item is outlined in FSAR Section 6.2.4, 7.3.1.1, and 7.3.1.3.2.

<u>ACCIDENT MONITORING INSTRUMENTATION (II.F.1)</u>

Necessary procedures required for this item will be prepared prior to fuel load at SHNPP. SHNPP compliance with the parts of this item concerning the Noble Gas Monitor, the Iodine and Particulate Monitor, and the Containment High Range Radiation Monitor is outlined in FSAR Sections 11.5.2 and 12.3.4 SHNPP compliance with the parts of this item concerning the Containment Pressure Monitor, the Containment Water Level Monitor and the Containment Hydrogen Monitor is outlined below.

Containment Pressure Monitoring - Continuous indication of containment pressure is provided in the control room. Redundant indicators whose inputs are derived from signals used in the Engineering Safety Feature Actuation System are provided on the main control board for a range of 0 to 55 psig. Another set of redundant indicators which monitor the effectiveness of the Containment Vacuum Relief System are provided on the auxiliary equipment panel having a range of ±5 inches of water column. Additionally, transmitters are provided with a range of -5 psig to 135 psig for containment pressure which will display on the SPDS display located on the main control board. See FSAR Section 6.2.1.1.2 for additional detail.

Containment Water Level Monitoring - Continuous monitoring of containment water level will be provided in the control room and displayed upon demand on

the SPDS display located on the main control board. Redundant instrumentation channels are provided to measure a narrow level range from the bottom of the containment sump (Elev 207') to the top of the sump (Elev 211'). Redundant wide range instrumentation channels are provided to measure water level from the bottom of the containment (Elev 211') to the elevation (Elev 228') equivalent to a 600,000 gallon capacity. Additionally, level transmitters are provided in each of the containment spray pump portions of the containment recirculation sumps with indication in the main control room. The containment recirculation sump besides being displayed on the SPDS, is also displayed on an indicator located on the main control board. The range covered by the sump transmitters is from near the bottom of the sump (Elev 216' 3") to the top of the sump (Elev 221' 3") which is sufficient to monitor the minimum water level for the containment spray pump NPSH requirements. See FSAR Section 6.2.1.1.2 for additional detail.

Containment Hydrogen Monitoring - Continuous indication of hydrogen concentration in the containment atmosphere is provided on the hydrogen analyzer remote control panel located adjacent to control room but within the overall control room complex environmentally controlled envelope. There are two redundant hydrogen analyzers with associated control panels which, therefore, provides redundant indication channels. In addition, the hydrogen concentrations will be displayed upon demand on the SPDS. See FSAR Section 6.2.5.2.3 for additional detail.

INSTRUMENTATION FOR DETECTION OF INADEQUATE CORE COOLING (II.F.2)

SHNPP will utilize procedures which provide adequate guidance to facilitate the operator's recognition of inadequate core cooling using instrumentation that will be available when the plant begins operations.

The Westinghouse Owners' Group, of which the Carolina Power & Light Company is a member, has performed analyses to study the effects of inadequate core cooling. These analyses were provided to the NRC "Bulletins and Orders Task Force" for review on October 31, 1979. As part of the submittal made by the Owners' Group, "Instruction to Restore Core Cooling During a Small LOCA" was included. This instruction provides the basis for procedure changes and operator training required to recognize the existence of inadequate core cooling and restore core cooling based on existing instrumentation. The SHNPP will incorporate the key considerations of this instruction into the procedure concerning incidents involving reactor coolant system depressurization and will provide training to the operators in this area. The key considerations which will be incorporated include: indications which are available to identify inadequate core cooling (subcooling monitoring, incore thermocouples, abnormal steam generator pressure, core differential temperature, and source or intermediate range nuclear instrumentation), methods to restore core cooling, and methods to eliminate a non-condensible gas bubble from the reactor vessel.

SHNPP will provide indication of the subcooling margin in the control room. Specific details of the system will be provided in a later amendment. Carolina Power & Light Company has evaluated and is continuing to evaluate the effectiveness of various instrumentation systems with which to detect and indicate inadequate core cooling through a display. Except for the subcooling monitoring described previously, no additional instrumentation has been identified which provides an unambiguous indication of inadequate core cooling or core uncovery.

Carolina Power & Light Company has done extensive investigation of reactor vessel level instrumentation. The designs presently available are neither reliable nor unambiguous. However, CP&L is continuing to work with potential vendors in an attempt to solve these drawbacks. As additional information becomes available CP&L will inform the NRC.

POWER SUPPLIES FOR PRESSURIZER RELIEF VALVES, BLOCK VALVES, AND LEVEL INDICATORS (II.G.1)

SHNPP compliance with this letter is outlined in FSAR Section 8.3.1.2.35.

IE BULLETINS – REVIEW ESF VALVES (II.K.1.5)

SHNPP procedures will address the positioning of ESF valves and will establish methods of demonstrating on a regular basis that these valves are in their proper position. These procedures will be completed and have been reviewed by the NRC staff prior to fuel load.

OPERABILITY STATUS (II.K.1.10)

Carolina Power & Light Company will review and modify, as necessary, SHNPP procedures for removing safety-related systems from service (and restoring to service) to assure operability status is known.

Procedures will govern the pre-maintenance and post-maintenance safety-related system status. Typically, to remove safety-related equipment from service, two operations documents may be initiated: 1) the Operations Work Procedure (OWP), or 2) the Equipment Inoperable Record (EIR).

An OWP is used in conjunction with an EIR when a specific lineup must be performed due to the component inoperability (such as tripping bistables or isolating containment ventilation), or if a tracking mechanism is needed to verify all compensatory actions are completed within the required times. It may also be used instead of initiating an EIR if all actions and time limits are listed. Double verification lineups are provided on the OWP for maintenance and normal positions if required.

An EIR logs equipment out of service and lists any time limitations that apply. EIRs and OWPs are reviewed at shift turnover to verify time limitations have been met.

Either the OWP or the EIR will list testing required on redundant equipment prior to and during maintenance and testing required to restore operability.

TRIP PER LOW-LEVEL BISTABLE (II.K.1.17)

SHNPP is a facility that does not use pressurizer water level coincident with pressurizer pressure for automatic initiation of safety injection. However, SHNPP does comply with the requirements intent by initiating safety injection upon a low pressurizer pressure signal. The details are described in FSAR Section 7.3.1.1 and Figure 7.3.1.1, Sheet 2 of 7.

THERMAL MECHANICAL REPORT (II.K.2.13)

This item required a detailed analysis of the thermal-mechanical conditions in the reactor vessel during recovery from small breaks with an extended loss of all feedwater. Westinghouse in support of the Westinghouse Owners' Group has performed the analysis for generic Westinghouse plant groupings to address this issue. The report on this analysis WCAP-10019, "Summary Report on Reactor Vessel Integrity for Westinghouse Operating Plants," was submitted to the NRC in December, 1981. This item is complete.

VOIDING IN RCS (II.K.2.17)

Westinghouse (in support of the Westinghouse Owners Group has performed a study which addresses the potential for void formation in Westinghouse designed nuclear steam supply systems during natural circulation cooldown/depressurization transients. This study has been submitted to the NRC by the Westinghouse Owners Group (letter 06-57, dated 4-20-81) and is applicable to SHNPP.

In addition, the Westinghouse Owners Group has developed a natural circulation cooldown guideline that takes the results of the study into account so as to preclude void formation in the upper head region during natural circulation cooldown/depressurization transients, and specifies those conditions under which upper head voiding may occur. These Westinghouse Owners Group generic guidelines have been submitted to the NRC (letter 06-64, dated 11-30-81). The generic guidance developed by the Westinghouse Owners Group (augmented as appropriate with plant specific consideration) will be utilized in the implementation of SHNPP plant specific operating procedures.

BENCHMARK ANALYSIS - SEQUENTIAL AFW FLOW (II.K.2.19)

Subsequent to the issuance of NUREG-0737, the NRC completed a generic review on this subject and concluded that the concerns expressed in Item II.K.2.19 are not applicable to NSSSs with inverted U-tube steam generators such as those designed by Westinghouse. Therefore, this item is not applicable to SHNPP and no further action is necessary.

AUTOMATIC PORV ISOLATION (II.K.3.1)

Based on the results of the Westinghouse Report, WCAP-9804, dated February 1981, Carolina Power & Light Company (CP&L) believes that no modifications at SHNPP are necessary. Carolina Power & Light Company considers this item complete.

REPORT ON PORV FAILURES (II.K.3.2)

The Westinghouse Owners' Group, of which Carolina Power & Light Company (CP&L) is a member, submitted WCAP-9804, "Probalistic Analysis and Operational Data in Response to NUREG-0737 Item II.K.3.2 for Westinghouse NSSS Plants" on March 13, 1981. WCAP-9804 summarizes the Westinghouse PWR owners' collective operational experience for pressurizer PORV and safety valves, and using this data in conjunction with industry-generic probabilistic data estimates the probability of a small-break LOCA caused by a stuck open PORV. With the submission of WCAP-9804 CP&L considers this item complete for SHNPP.

REPORT ON SAFETY AND RELIEF VALVE FAILURES AND CHALLENGES (II.K.3.3)

At SHNPP, any failure of a relief or safety valve will be reported to NRC in accordance with plant LER procedures.

AUTOMATIC TRIP OF REACTOR COOLANT PUMPS (II.K.3.5)

Westinghouse (in support of the Westinghouse Owners Group) has performed an analysis of delayed reactor coolant pump trip during small-break LOCAs. This analysis is documented in WCAP-9584, dated August 1979. In addition, Westinghouse (again in support of the Westinghouse Owners Group) has performed test predictions of LOFT Experiments L3-1 and L3-6. The results of these predictions are documented in the following Owners Group letters to the NRC: 06-49, dated 3-3-81; 06-50, dated 3-23-82; 06-60, dated 6-15-81. Based on: 1) the Westinghouse analysis, 2) the excellent prediction of the LOFT Experiment L3-6 results using the Westinghouse analytical model, and 3) Westinghouse simulator data related to operator response time, the Westinghouse and CP&L position is that automatic reactor coolant pump trip is not necessary since sufficient time is available for manual tripping of the pumps.

Our understanding of the schedule for final resolution of this issue is:

a) Once the NRC formally approves the Westinghouse model, a 3-month study period will ensue during which the Westinghouse Owners Group will attempt to demonstrate compliance with some NRC acceptance criteria for manual RCP trip. The NRC acceptance criteria will accompany their formal approval of the Westinghouse models.

b) If, at the end of the 3-month period, the Westinghouse Owners Group cannot show compliance with the acceptance criteria, the NRC will formally notify utilities that they must submit an automatic RCP trip design.

EVALUATION OF PORV OPENING PROBABILITY (II.K.3.7)

This item is not applicable to SHNPP.

PID CONTROLLER (II.K.3.9)

The SHNPP Proportional Integral Derivative Controller has the derivative action setting set to zero, thereby eliminating it from consideration. No modifications are required for this item. CP&L considers this item complete.

APPLICANT'S PROPOSE ANTICIPATORY TRIP AT HIGH POWER (II.K.3.10)

No modification to the standard anticipatory trip has been proposed. This item is not applicable to SHNPP.

JUSTIFY USE OF CERTAIN PORV'S (II.K.3.11)

This item is not applicable to SHNPP.

CONFIRM ANTICIPATORY TRIP (II.K.3.12)

SHNPP compliance with this item is outlined in FSAR Section 7.2.1.

FINAL RECOMMENDATIONS, B & O TASK FORCE - ECCS OUTAGES (II.K.3.17)

In the first five years following issuance of an operating license for SHNPP, Unit 1, CP&L will collect the following data on ECC systems and components:

a) Outage dates and duration of outages.

b) Cause of each outage.

c) ECC systems or components involved in the outage.

d) If applicable, corrective actions required.

Test and maintenance outages will be included in the data collection. A report of the data collected will be submitted to the NRC within one year of the end of the five-year period.

EFFECT OF LOSS OF AC POWER ON PUMP SEALS (II.K.3.25)

This item requires that the consequences of a loss of RCP seal cooling due to a loss of AC power (defined as loss of offsite power) for at least 2 hours be demonstrated.

During normal operation, seal injection flow from the chemical and volume control system is provided to cool the RCP seals and the component cooling water system provides flow to the thermal barrier heat exchanger to limit the heat transfer from the reactor coolant to the RCP internals. In the event of loss of offsite power the RCP motor is de-energized and both of these cooling supplies are terminated; however, the diesel generators are automatically started and either seal injection flow or component cooling water to the thermal barrier heat exchanger is automatically restored within seconds. Either of these cooling supplies is adequate to provide seal cooling and prevent seal failure due to loss of seal cooling during a loss of offsite power for at least 2 hours.

In the event of a loss of onsite AC power (due to a fire) which causes a loss of seal cooling, the Alternate Seal Injection System will automatically actuate after a time delay and restore seal cooling within 3 minutes of initial loss of seal cooling flow.

Based on this CP&L believes no modifications at SHNPP are necessary in response to item. CP&L considers this item as complete.

REVISED SB LOCA METHODS TO SHOW COMPLIANCE WITH 10CFR50, APPENDIX K (II.K.3.30)

The Siemens SB LOCA evaluation model was used to analyze the SB LOCA as summarized in Section 15.6.5. The model consists of the following:

1) XN-NF-82-49(A), Revision 1, "Exxon Nuclear Company Evaluation Model - EXEM PWR Small Break Model," June 1986.

2) XN-NF-82-49(P)(A), Revision 1, Supplement 1 and Correspondence, "Exxon Nuclear Company Evaluation Model - EXEM PWR Small Break Model," December 1994.

3) XN-NF-81-58(A), Revision 2, and Supplements 1 through 4, "RODEX 2 Fuel Rod Thermal - Mechanical Response Evaluation Model," Revision 2 and Supplements 1 and 2 dated March 1984, Revision 2, Supplements 3 and 4 dated June 1990.

PLANT SPECIFIC CALCULATIONS TO SHOW COMPLIANCE WITH 10CFR50.46 (II.K.3.31)

The discussion and results of the small break LOCA analysis using the Siemens SB LOCA evaluation model has been incorporated in FSAR Section 15.6.5.

In accordance with the NRC Generic Letter 83-35 (November 2, 1983) this analysis demonstrates compliance with 10CFR50.46.

EMERGENCY PREPAREDNESS, SHORT TERM (III.A.1.1)

Refer to the SHNPP Emergency Plan. The pre-licensing exercise for SHNPP is scheduled for the fall of 1984. Meteorological data collection for emergencies is described in Section 3, SHNPP Emergency Plan.

UPGRADE EMERGENCY SUPPORT FACILITIES (III.A.1.2)

Refer to Section 3, SHNPP Emergency Plan.

PRIMARY COOLANT OUTSIDE CONTAINMENT (III.D.1.1)

A program designed to reduce leakage from systems outside Containment that would contain highly radioactive fluids during a serious transient or accident will be implemented. This program will be initiated during the preoperational test phase. The program will be performed per approved procedures by qualified personnel.

The program is designed to detect leakage to RAB atmosphere from systems which would be used to bring the plant to a safe shutdown following a serious transient or accident.

a) Systems Included in the Leak Reduction Program

   A review of the plant design has resulted in the following systems being included in the leak reduction program.

   1) Residual Heat Removal System

   2) Safety Injection System, except boron injection recirculation subsystem

   3) Containment Spray System, except spray additive subsystem and RWST

   4) Chemical & Volume Control System

      a. Letdown Subsystem

      b. Boron Recycle Hold-Up Tanks (RHT)

      c. Charging Pumps

   5) Post-Accident Sample System

   6) Post-Accident RAB Ventilation System

7) Valve Leakoff Equipment Drain System

8) Gaseous Waste Processing System

9) Seal Water Return System

10) Demineralizer in the Chemical and Volume Control System

11) The portion of the Filter Backwash System that services the A and B RCP Seal Injection Filters

12) The Post-Accident Hydrogen Monitoring System

b) System Excluded from the Leak Reduction Program

The following systems have been excluded from the leak reduction program because they will not be used to mitigate the consequences of an accident:

1) Chemical and Volume Control

a. Boron Thermal Regeneration System

b. Boric Acid Transfer System

2) Filter Backwash System

3) Fuel Pool Cooling and Cleanup System

4) Radioactive Waste Disposal Systems

5) Process Sampling System

c) Explanation for Exclusion of Systems from Leak Reduction Program

1) Chemical and Volume Control

a. The Boron Thermal Regeneration System (BTRS) will not be tested for the following reasons:

1) The BTRS will not be used in a post-accident situation.

2) The BTRS has no functional design characteristics which can be used for post-accident mitigation.

3) The BTRS was not included in the list of systems requiring testing by NUREG 0737.

b. The Boric Acid Transfer System will not be used, because boration will be provided by adding borated water from the Refueling Water Storage Tank while letting down to the Recycle Holdup Tank.

c. The only component within the Boron Recycle System that may be used post-accident is the Recycle Holdup Tank (RHT). This will allow a letdown path to be established for reactor coolant in order to provide the volume reduction required to permit boration. The feed pumps and feed filters will not be required post-accident, because total letdown for boration would be adequately held by the 84,000 gallon RHT during the post-accident phase. Treatment of the effluent in the RHT will not occur until the recovery phase, well after the post-accident phase. Actions at that time can be thoroughly planned and deliberate. Therefore, although the RHT will be included in the leak reduction program, the balance of the system need not be.

2) Filter Backwash System

The filter backwash system is designed to collect materials back flushed from the filters within systems such as the chemical and volume control system. Filters which are part of systems included in the leak reduction program will be included in the boundary subject to the leak reduction program. Such filters are not in service during safety injection or recirculation and can be manually bypassed in the event the pressure drop becomes too great or during flushing operations. The back flushing portions of the system are operated manually and the back flushing alignment exists for only short periods of time. Backflushing is not likely to occur following an accident since it would only tend to spread contamination. Therefore, CP&L concludes that this system does not need to be included in the leak reduction program.

None of these systems are required by NUREG 0737 Item III.D.1.1 to be included in the leak reduction program, and CP&L believes that no significant benefit would be derived from their inclusion.

3) Fuel Pool Cooling and Cleanup System

The SER stated that the Fuel Pool Cooling and Cleanup system should be leak tested to minimize leakage that might occur during a serious transient or accident (i.e. fuel handling accident). The Fuel Pool Cooling and Cleanup System is used to maintain water quality in the fuel pools and as such can reduce the contamination present in the fuel pools. However, the fuel pools do not have a significant inventory of radioactive material following a fuel handling accident since all noble gases are assumed to escape from the pool surface. This postulated accident, even when conservatively analyzed contributes only .02 microcuries of iodine per gram to the activity of the water. CP&L does not consider this to represent a highly contaminated system, especially within the context of TMI Item III.D.1.1.

We concluded that this system need not be included in the leak reduction program.

4) Radioactive Waste Disposal

5) Process Sampling System

The Process Sampling System is not used for post-accident sampling. In accordance with NUREG-0737, Item II.B.3, a separate Post-Accident Sampling System (PASS) is used. A description of the PASS is found in FSAR Section 9.3.2.2.3. The PASS will be leak tested as provided in our commitment for testing.

Carolina Power & Light Company has evaluated potential leak paths as discussed in NRC letter regarding the North Anna incident dated October 17, 1979. Design changes will be implemented, if need is identified, prior to fuel load to eliminate the open paths to the atmosphere as discussed in the letter.

The leak reduction program employs visual inspections of the mechanical joints and seals of the system to detect leakage and measurement of observed leakage. These inspections will be conducted with the system pressurized to normal operating pressure using the system fluid as a test medium. The observed leakage will be documented and compared against the acceptance criteria. Corrective action will be taken as appropriate, but in all cases will be to reduce leakage to as-low-as-practical levels. Testing of gaseous systems will include helium leak detection or equivalent testing methods. The initial leak test submittal will be made two months prior to fuel load.

After commercial operation a program of preventative maintenance to reduce leakage to as-low-as-practical levels will be established and implemented. The program will include periodic leak tests at each refueling cycle interval or less.

INPLANT IODINE RADIATION MONITORING (III.D.3.3)

CP&L Response to NRC Position and Clarification

The Shearon Harris Nuclear Power Plant will comply with the requirement to accurately detect the presence of iodine in regions of interest following an accident in the following manner.

The SHNPP will have the capability to analyze for airborne iodine in a counting laboratory which has a low background, low contamination and will be ventilated with clean air. This capability includes the ability to analyze effluent charcoal samples. If the counting laboratory background and contamination levels exceed established limits, the counting facilities of the Harris Energy and Environmental Center and other CP&L nuclear facilities will be utilized.

CONTROL ROOM HABITABILITY (III.D.3.4)

SHNPP compliance with this item is outlined in FSAR Section 6.4 as augmented by FSAR Section 9.4.1.

Question No.	Response Provided in FSAR Section No.	Amendment No.
210.01	Table 3.2.1-1	9
210.02	Table 3.2.1-1	9
210.03		No Revision
210.04		No Revision
210.05		No Revision
210.06	3.6.1.3	9
210.07	3.6.2.1.1.2	
210.08	3.6.2.1.1.2	9
210.09	3.6.2.1.1.3	9
210.10	3.6.2.1.4, 6.6.8, Figure 3.6.2-1	9
210.11	3.6.2.1.4, 6.6.8, Figure 3.6.2-1	9
210.12	3.6.2.2.1, Figures 3.6A-1 through 3.6A-27	9
210.13		No Revision
210.14		No Revision
210.15	3.6.2.3.4.2	9
210.16	3.6.2.5.1.a	
210.17		No Revision
210.18		No Revision
210.19	Table 3.6.2-2	9
210.20	Tables 3.6A-15, -16, -17.1, -17.2, -17.3, and -18, Figure 3.6A-1	9
210.21		No Revision
210.22	3.7.3	19
210.23	3.7.3.1.1	9
210.24		No Revision
210.25	3.7.3.8.1, Figure 3.7.3-1	9
210.26		No Revision
210.27	3.7.3.1.1, 3.7.3.9.1, 3.7.3.9.2, 3.7.3.13	9
210.28	Tables 3.9.1-2, 3.9.1-3	9
210.29	3.9.1.2.2	9
210.30	3.9.1.2.2	9
210.31	3.9.1.4.7	9
210.32	2.12.1.12, 3.9.2.1.1, 3.9.2.1.3, 14.2.12.1.12	9 and 17
210.33	3.7.3.1.1, 3.7.3.9.1, 3.7.3.9.2, 3.7.3.13	9 and Future

Question No.	Response Provided in FSAR Section No.	Amendment No.
210.34	1.8 (RG 1.48)	9
210.35	3.9.3.1, 3.9.3.1.1, 3.9.3.1.2.2, Table 3.9.3-7	9 and 17
210.36		No Revision
210.37		No Revision
210.38	Tables 3.9.3-10 and -11	9
210.39	3.9A	9
210.40	3.9.3.4.2	
210.41		No Revision
210.42	Table 3.9.3-1	9
210.43		No Revision
210.44	3.9.6.2	9 and 11
210.45	Appendix 3.9D	No Revision
220.02	3.8.4.9, All Chapters	5
220.03	3.3.1	5
220.4	3.5.3.2.1	5
220.5	3.5.3.1.3, Reference 3.5	5
220.6	3.8.3.7, 3.8.4.7 and 3.8.5.7	No Revision
220.7	3.7.2.4A, and 3.7.2.5A	No Revision
220.8	3.7.2.4A	3
220.9	3.7.2.15A, Reference 3.7	5
220.10	3.7.2.1A	5
220.11	3.7.2.1A, Reference 3.7.2	5
220.12	3.7.2.15A, (3.7.2.4A No Revision)	5
220.13	3.7.2.1A	5
220.14	3.7.2.1A, and Figures 3.7.2-13 to -16	5
220.15	3.7.2.4A, and Figure 3.7.2-8	5
220.16	3.7.2.4A	5
	3.7.2.16B	14
220.17	3.7.2.4A	5
	3.7.2.16B	14
220.18	3.7.3.1.1.1, Table 3.7.3-2A and Figure 3.7.3-12	5
	3.7.3.1.1.1(b)	14
220.19	3.8.1.4.4.4, and Figure 3.8.1-28	5
220.20	Question Deferred	*
220.21	3.8.3.1.6, and Figure 3.8.3-16 (3.8.3 No Revision)	5

Question No.	Response Provided in FSAR Section No.	Amendment No.
220.22	3.8.1.4.4.9	5
220.23	3.8.1.1.3.2	14
220.24	3.8.4.1.14, and Table 3.8.4-3	5
220.25	3.8.4.8	5
220.26	3.8.4.3.1, 3.8.5.5, and Reference 3.8	5
220.27	3.8B.6	5
230.1	Table 2.5.2-1	5
230.2	Figure 2.5.2-1a, and Figure 2.5.2-17	5
230.3	2.5.2.1	No Revision
230.4	2.5.2.6, Figure 2.5.2-18	5
231.2	2.5.0.3.2, 2.5.0.3.3, and 2.5.1.2.3	5
231.3	2.5.3.3	1
231.4	Table 2.5E-1	5
231.5	Table 2.5E-2	5
231.6	Table 2.5E-3	5
231.7	Figures 3 & 4 of Section 2.5E	Future
231.8	Figs. 5 & 6 of Sec. 2.5E (Fig. 2 sheet 9 not completed yet)	Future
240.6	2.51	14
240.7	Figure 2.4.2-5	10 and 14
240.10	2.4.8	14
240.11	2.4.1.1, 2.4.8	14
252.1	4.5.2.1, 5.2.3.4	17
252.2	4.5.1.1, 4.5.2.1, Table 4.5.1-1	17
252.3		No Revision
252.4	1.8 (RG 1.34)	17
252.5	5.2.3.1, 5.4.10.2.2	17
252.6		No Revision
252.7	10.3.6.1	17
260.67	5.2.1.2, Tables 3.2.1-1 and 5.2.1-2	14
280.1		No Revision
280.2	9.5.1, and Tables 9.5.1-6, -7, -8	5
280.3	9.5.1.5	5
280.4	9.5.1.5	5
280.5	9.5.1.5	5
260.6	9.5.1.2.2, and Figures 9.5A-1 thru 9.5A-40	5
280.7	9.5.1.2.2 (Page 9.5.1-9)	No Revision

Question No.	Response Provided in FSAR Section No.	Amendment No.
280.8	9.5.1.2.2	5
280.9	9.5.1.2.2	
280.10	9.5.1.2.2	5
280.11	9.5.1.2.2	14
280.12	9.5.1.2.2	5
280.16	9.5.1.5	5
280.19	9.5.2.2.1	5
280.20	9.5.1.1.3	5
280.21	4.3.2.4	5
280.24	9.5.1.2.3	14
280.25	9.5.1.2.3 (Page 9.5.1-26)	No Revision
280.26	9.5.1.2.3 (Page 9.5.1-19)	No Revision
280.27	9.5.1.2.3	5
280.28	already in 9.5.1.2.3 (Page 9.5.1-26)	5
280.29	9.5.1.2.4	5
280.30		No Revision
280.31	9.5A.11 (9), 7.4.1.11	5
280.32	9.5.1.2.4 (Page 9.5.1-30)	No Revision
280.33	9.5.1.2.4	5
280.34	9.5.1.2.3 (Pages 9.5.1-22 & -23)	No Revision
280.35	9.5.1.2.4 (9)	5
280.36	9.5.1.2.4	5
281.1	1.8 (Page 1.8-67)	5
281.2	6.1.2.1, and Table 6.1.2-2	5
281.3	9.1.3.2	5
281.4	9.3.2.1, and 9.3.2.2.3	5
282.1	9.1.3.2, and Table 9.1.3-3	5
282.2	10.3.5, Tables 10.3.5-2, -3, -4, & -5 and Section 10.3.5.4	5
311.6	1.8 (Page 1.8-108), and 2.2.3	No Revision
	2.2.3.3	14
311.7	2.1.3.3, 2.1.3.4, and Table 2.1.3-5	5
410.2	3.5.1.1.2, and Table 3.5.1-17	5
410.3	3.5.1.1.2	5
410.4	3.5.1.2.1.2	5
410.5	3.6.2.1.2, 3.6.2.1.4, 3.6A.3.2, 3.6.2.1.2.1, and Figures 3.6A-34 to -42	5
410.6	9.4.8.3	5

Question No.	Response Provided in FSAR Section No.	Amendment No.
410.7	5.2.5.3.1, 5.2.5.6.2, and 5.2.5.8.2	1
410.8	9.1.1, 9.1.2, 9.1.3, Tables 9.1.3-1 and -9, Figures 9.1.1-1 and -2	7 and 14
410.9	9.1.3.3	5
410.10	15.7.4, and 9.1.2.3	No Revision
410.11(1)	9.1.4.2.2.7, Figure 9.1.4-7	14
410.11(2)	9.1.4.2.2.7, Figure 9.1.4-7	14
410.11(3)	9.1.4.2.2.7, 9.1.4.2.2.8, Appendix 9.1A	14
410.12	9.1.4.2.2.7, 9.1.4.2.2.4	5
410.13	(Analysis Available in 1983)	*
	9.2.2.3, 9.2.2.5	14
410.14	9.2.6.3 and Figure 9.2.6-1	5
410.15	9.3.1.2, 9.3.1.3 and Table 9.3.1-2	5
410.16	9.3.1.2	5
410.17	9.3.3.3	5
410.18	10.4.5.3	5
410.19	14.2.12.2.31	No Revision
410.20	10.4.9.1	5
410.21	Figure 10.1.0-1 and 10.1.0-3	5
410.22(1)	Table 10.4.9-3, Section 10.4.9.3	14
410.22(2)	Appendix 10.4.9.A	14
410.22(3)	9.2.6.5, 10.4.9.3	14
410.22(4)	10.4.9.1, 15.2.6, 10.4.9.13	14
410.22(5)	7.3.1.9, 7.3.1.9a, 7.3.1.9b, 7.3.1.10, 10.1.0.3	5, 9, and 14
420.5	8.3.1.1.1.4, Table 8.3.1-9	17
420.7	7.7.2.1	Future
420.8	7.7.2.1	Future
420.9	7.2.2.3.5	Future
430.12	8.3.1.1.1.5 and 13.2.1.1.1	5
	8.3.1.1.1.5	17
430.13	1.8 (RG 1.108)	17
430.14	8.3.1.1.2.14	5
430.15	9.5.2.2	No Revision
	9.5.2.2.1	17
430.16	9.5.2.2.1	5
430.17	9.5.3.2	5
	9.5.3.3, Table 9.5.3-1, Figures 9.5.3-1 and -2	17
430.18	9.5.3.3	5
	9.5.3.3, Table 9.5.3-1, Figures 9.5.3-1 and -2	17
430.19	No Revision Required	

Question No.	Response Provided in FSAR Section No.	Amendment No.
430.20	No Revision Required	
430.21	9.5.4.3	5 and 17
430.22	9.5.4.3 c) and 9.5.4.5	5
	9.5.4.3	17
430.23	9.5.4.2	17
430.24	9.5.4.3	5 and 17
430.25	9.5.4.3	5
430.26	9.5.4.6	5 and 17
430.27	9.5.4.1	5 (Figures will be sent later)
	9.5.4.1, 9.5.4.2, 9.5.4.3, 9.5.4.4, 9.5.5.3	17 (Figures will be sent later)
430.28	9.5.4	5
	1.8, 9.5.4.5	8
430.29	9.5.4.3	5
430.30	Table 3.2.1-1	5
430.31	9.5.4.3	5
	9.5.4.3, Figure 9.5.4-2	17
430.32	9.5.4.3 and Figures 9.5.4-1, 9.5.4-2, and 9.5.5-2	5
	9.5.4.3, Figure 9.5.4-2	17
430.33	9.5.4.2 and Figure 9.5.5-2	5
430.34	9.5.4.3	5
	9.5.4.1, 9.5.4.3	17
430.35	3.6.2.1.7	5
	9.5.5.3	17
430.36	Future	5
	9.5.4.2, 9.5.4.3, Figure 9.5.4-2	17
430.37	9.5.5.2	5
430.38	9.5.5.2	5
430.39	9.5.5.4 and 9.5.5.5	5
430.40	9.5.5.5 and Figure 9.5.5-1	5
430.41	9.5.5.5 and Figure 9.5.5-1	5
430.42	8.3.1.1.2.14	5
	8.3.1	17
430.43	9.5.5.2	5
430.44	9.5.5.3	5
430.45	Figure 9.5.5-1	5
430.46	9.5.5.2	5
430.47	9.5.5.3, Table 3.2.1-1	5
	9.5.5.3, Figure 9.5.5-1	17
430.48	9.5.6.3 and 9.5.6.4	5
430.49	9.5.6.3	5
	9.5.5.3, Figure 9.5.5-1	17

Question No.	Response Provided in FSAR Section No.	Amendment No.
430.50	9.5.6.4	5
	9.5.4.5, 9.5.5.4, 9.5.6.4, 9.5.7.4, 9.5.8.4	17
430.51	Figure 9.5.6-1	5
	9.5.6.3, Table 3.2.1-1, Figure 9.5.6-1	17
430.52	9.5.7.2	5
430.53	9.5.7.2	5
430.54	9.5.7.4 and 9.5.7.3	5
430.55	9.5.7.4 and Table 9.5.7-2	5
430.56	9.5.7.2	5
430.57	9.5.7.3	5 and 17
430.58	9.5.7.2, Table 3.2.1-1	5
	9.5.7.2	17
430.59	9.5.8.4 and 9.5.8.5	5
430.60	9.5.8.3	5
430.61	9.5.8.3	5
430.62	9.5.8.3, 9.4.5.2.5.1 and 9.4.5.2.5.2	5
430.63	9.5.8.3, Table 3.2.1-1 and Figure 9.5.5-2	5
	9.5.8.3	17
430.64	9.5.8.2, and Figure 9.5.5-2	5
	9.5.8.3	17
430.65	10.2.2, and Figures 10.2.2-7, -8, -9, -10	5
	10.2.2	17
430.66	10.2.2, (10.2.3 and 3.5.1.3 No Revision)	5
430.67	10.2.2	5
430.68	10.2.3.6	No Revision
430.69	10.2.3.6	17
430.70	10.2.4	5 and 17
430.71	10.2.2	5
430.72	10.2.2	5
	9.3.9, 9.3.4.1.1.2	17
430.73	10.3.2, 10.3.3, and Table 10.3.2-1	5
430.74	10.4.1.2	5
430.75	10.4.1.3	5
430.76	10.4.1.2	5
430.77	10.4.1.1	5
430.78	10.4.1.5 and Table 10.4.1-2	5
	10.4.1.5	17
430.79	Table 10.4.1-2 (10.4.1.2 No Revision)	5
430.80	10.4.2.2, and 11.5.2.7.2.9	5
430.81	10.4.1.3	5
430.82	10.4.1.4	5
430.83	Figure 7.2.1-1 Sheet 10, Figure 7.7.1-8, 10.4.4 and Table 10.4.4-2	No Revision

Question No.	Response Provided in FSAR Section No.	Amendment No.
430.84	Figures 7.2.1-1 and 10.1.0-1	No Revision
430.85	10.4.4.4	17
430.86	10.4.4.3	5
	10.2.4	17
430.87	10.4.4.3 and Table 10.4.4-2	No Revision
430.89	8.2.1.1, Figures 8.2.1-10, -11, and -12	5
	8.3.1.1.2.4	17
430.91	8.2.2.3	5
430.92	8.3.1.2.19, and 1.8	5
430.93	8.3.1.1.2.11	5
	8.3.1.1.2.1, 8.3.1.1.2.3, 8.3.1.1.2.5, 8.3.1.1.2.11, 8.3.1.1.2.15, 8.3.1.1.3, 14.2.12.82	17
430.94	8.3.1.1.2.8	5
430.95	8.3.1.1.2.14 j)	5
430.96	8.3.1.2.4	5
430.97	8.3.1.2.38	5
	6.3.1, 8.3.1.2.38	17
430.98	3.1, 3.10, 3.11 and 8.3.1	No Revision
430.99	8.3.1.1.2.8	5
430.101	8.3.1.1.2.14 f), Table 8.3.1-2, Figure 8.3.1-12 and -13	Future
430.102	8.3.1.1.2.14k, and Table 8.3.1-8	5
	8.3.1.1.2.14k	17
430.103	8.3.1.1.2.17	5
430.104	8.3.1.3	5
430.105	1.8 and 8.3.1.2.14	5
	8.3.1.1.2.5, 8.3.1.2.14	17
430.106	8.3.1.1.2.4, 5.4.7.2.6 and Figure 8.3.1-5	5
	8.3.2.1.2, 8.3.2.1.3	17
430.107	8.3.1.2.35	
430.108	8.3.1.1.2.11	5
430.109	8.3.1.1.1	5
430.110	7.3.1.3.3	5
430.111	8.3.1.1.2.14 g)	5
430.112	8.3.1.1.2.4	5
430.113	8.3.1.1.2.15, 8.3.1.2.11, and Figures 8.3.1-6 to -11	5
	8.3.1.1.2.15 f), 8.3.2.1.3	17
430.114	8.3.1.1.3	5
430.115	8.3.2.1.1	5
430.116	8.3.2, and Tables 8.3.2-1, -2, -3, and -4	5
430.117	8.3.2.1.2, 8.3.2.1.1, and 8.3.2.1.4	5
430.118	8.3.2.2.1.3	5
430.119	8.3.2.1.2	5

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640.5.5.z	14.2.12.1.14	No Revision
640.5.5.a.a	14.2.12	No Revision
640.5.5.b.b	14.2.12.2.28	No Revision
640.5.5.e.e	14.2.12.1.57	No Revision
640.5.5.k.k	14.2.12.2.29, 15.1	4
640.5.5.m.m	14.2.12.1.50, 14.2.12.2.30, 15.2.3.2	4
640.6	14.2.12.2.20	No Revision
640.7	1.8, 14.2.7, 14.2.12.1.79	No Revision
640.8	14.2.7, 14.2.12.1.16	4
640.9		No Revision
640.10	14.2.12.1.6	4
640.11	14.2.12.1.28, 14.2.12.1.29	No Revision
640.12	1.8, 9.4.1.4, 14.2.7	4 and 7
640.13	14.2.10.1	4
640.14		No Revision
640.15	14.2.12.1.66	No Revision
640.16		4
640.17	14.2.12.2	4
640.18		No Revision
640.19(01)	14.2.12.1.1, 14.2.12.1.62, 14.2.12.2.5	4
640.19(02)	14.2.12.1.44, 14.2.12.1.63, 14.2.12.2.19	4
640.19(03)	14.2.12.1.25, 14.2.12.1.37, 14.2.12.1.61, 14.2.12.2.2	4
640.19(04)	14.2.12.1.16, 14.2.12.1.51, 14.2.12.1.52	4
640.19(05)	14.2.12.1.16	4
640.19(06)	14.2.12.1.3, 14.2.12.1.4, 14.2.12.1.6, 14.2.12.1.36, 14.2.12.1.50	4
640.19(07)	14.2.12.1.24, 14.2.12.1.25, 14.2.12.1.44, 14.2.12.1.49	4
640.19(08)	14.2.12.1.3, 14.2.12.1.4, 14.2.12.1.26, 14.2.12.2.1	4
640.19(09)	14.2.12.1.6	4
640.19(10)	14.2.12.1.60, 14.2.12.2.7, 14.2.12.2.11	4
640.19(11)	14.2.12.1.13, 14.2.12.1.50	4
640.19(12)	14.2.12.1.5, 14.2.12.1.6	4
640.19(13)	14.2.12.1.8, 14.2.12.1.25, 14.2.12.1.63	4
640.19(14)	14.2.12.1.63	4
640.19(15)	14.2.12.1.8	4
640.19(16)	14.2.12.1.45	4
640.19(17)	14.2.12.1.47, 14.2.12.1.66	4

Question No.	Response Provided in FSAR Section No.	Amendment No.
640.20(1)	14.2.12.1.4	4
640.20(2)	14.2.12.1.14	2 and 7
640.20(3)	14.2.12.1.20	4
640.20(4)	14.2.12.1.28	4
640.20(5)	14.2.12.1.37	4
640.20(6)	14.2.12.1.51	4
640.20(7)	14.2.12.1.66	4
640.21	14.2.12.1.25, 14.2.12.1.28, 14.2.12.1.29	No Revision
640.22	14.2.12.1.58	7
640.23	14.2.12	No Revision
640.24	14.2.12	No Revision
640.25	14.2.12	No Revision
640.26	14.2.12.1.2, 14.2.12.1.3.a(1)	No Revision
640.27	14.2.12.1.6	4
640.28	14.2.12.1.11, 14.2.12.2.19, 15.0.6	No Revision
640.29	14.2.12.1	4
640.30	14.2.12.1.39, 14.2.12.1.40	4
640.31	14.2.12.1.44	4
640.32	15.1.4.2	14
640.33(1)	14.2.12.2	No Revision
640.33(2)	14.2.12.2	No Revision
640.33(3)	14.2.12.2	No Revision
640.33(4)	14.2.12.2	No Revision
640.33(5)	14.2.12.2	No Revision
640.33(6)	14.2.12.2	No Revision
640.34	14.2.12.2.1	4
640.35	14.2.12.2.10	4
640.36	14.2.12.2.17	4
640.37	14.2.12.2.18	4
640.38(1)	14.2.12.2	No Revision
640.38(2)	14.2.12.2	No Revision
640.39	14.2.12.2.22	4

ACRONYMS USED IN APPENDIX B (NRC BRANCHES)

AEB	Accident Evaluation Branch
ASB	Auxiliary Systems Branch
CHEB	Chemical Engineering Branch
CPB	Core Performance Branch
CSB	Containment Systems Branch
EHEB	Environmental and Hydrologic Engineering Branch
EPB	Emergency Planning Branch
EQB	Equipment Qualification Branch
GSB	Geosciences Branch
HFEB	Human Factors Engineering Branch
ICSB	Instrumentation and Control Systems Branch
LQB	Licensing Qualification Branch
MEB	Mechanical Engineering Branch
METB	Meteorological and Effluents Treatment Branch
MTEB	Materials Engineering Branch
PSB	Power Systems Branch
PSRB	Procedures and Systems Review Branch
QAB	Quality Assurance Branch
RAB	Radiological Assessment Branch
RSB	Reactor Systems Branch
SAB	Siting Analysis Branch
SEB	Structural Engineering Branch
SGEB	Structural and Geotechnical Engineering Branch

DSER Open Item No.	NRC Branch	Subject (DSER Section)	Safety Review Question	Affected FSAR Section(s)	Amendment
1	SEB	The lumped mass-spring model approach and the finite boundary approach of analyzing the effect of soil/rock structure interaction should both be used. (3.7.3)	220.16, 220.17	3.7.2.16B	14
2	SEB	Input motion of multiple fixed branches and looped systems is transmitted to the piping system through the piping supports. (3.7.3)	210.18	3.7.3.1.1.1(b)	14
3	SEB	Delay of the ultimate capacity analysis of the containment. (3.8.1)	220.20	3.8.1	No FSAR Revision
4	SEB	Information regarding linear design and behavior of the dome and cylinder (thin-walled shells). (3.8.1)	220.23	3.8.1.1.3.2	14
5	SEB	The effect of cancelling Units 3 and 4 on Units 1 and 2 from a structural design point of view. (3.8)	220.02	All Chapters	10
6	SEB	Resolution of noncompliance with ASME Code Section III. (3.8)	---	---	---
7	SAB	Information needed on frequencies and quantities of shipment of toxic materials, shipped on the roads and railroads in the vicinity of the plant. (2.2.1)	311.6	2.2.3.3.1, 2.2.3.3.2	14
8	METB	Determining whether safety structures are capable of resisting 100-year snow pack and PMWP (48 hr.). (2.3.1)	---	3.8.4.3.1	17
9	METB	Operating basis condition for Category I structures (2.3.1)	---	3.8.1.3.1, 3.8.4.3.1, 3.8.4.3.2, 3.7.4.3.2, Table 3.8.1-2	---
10	METB	Meteorological program will be reviewed in reference to emergency preparedness. (2.3.3)	---	2.3.3.1	17
11	EHEB	A grading plan to be required regarding possible ponding during PMP. (2.4.2)	---	2.4.2.3, Table 2.4.2-4	14
12	EHEB	An analysis of the ponding of water on roofs during a local PMP. (2.4.2)	240.7	Figure 2.4.2-5	10 & 14

DSER Open Item No.	NRC Branch	Subject (DSER Section)	Safety Review Question	Affected FSAR Section(s)	Amendment
13	EHEB	An adequately designed riprap must be provided on the downstream face of the main dam. (2.4.3)	240.6	2.51	14
14	EHEB	Design a sediment-monitoring program. (2.4.11)	240.11	2.4.1.1, 2.4.8	14
15	EHEB	Inspection program to be established for safety-related water control structures. (2.4.11)	240.10	2.4.8	14
16	EHEB	Analysis of the augmentation from the Cape Fear River needed for the one unit and two unit operation. (2.4.11)	240.4, 240.12, 240.14	---	---
17	GSB	Complete investigation of the fault that follows the Neuse River. (2.5)	---	---	---
18	GSB	Resolve adequacy of the design-basis earthquake with respect to USGA position. (2.5.1.1)	---	---	---
19	SGEB	Summaries of the field test data to verify support of Category I pipes and conduits. (2.5.4.3)	---	---	---
20	SGEB	Summaries of field test to confirm that the backfill meet specifications for material type, density, and moisture content. (2.5.4.3)	---	---	---
21	SGEB	Construction notes that discuss changes in design details and construction procedures. (2.5.4.7)	---	---	---
22	SGEB	The applicant to describe plans and criteria for future monitoring of structure movements and piezometric levels. (2.5.4.8)	---	---	---
23	SGEB	The inspection and testing of residual soils in the auxiliary separating dike formation to be documented. (2.5.6.2.3)	---	---	---
24	SGEB	Summaries of the test data of the fill along the EMS intake channel alignment to met specifications. (2.5.6.2.3)	---	---	---
25	SGEB	To update summaries of field test results to include data obtained since December 1979 and to identify any results when specifications were not met. (2.5.6.3.6)	---	---	---

DSER Open Item No.	NRC Branch	Subject (DSER Section)	Safety Review Question	Affected FSAR Section(s)	Amendment
26	SGEB	Review the instrumentation program and proposed inservice inspection program for water control structures. (2.5.6.7)	---	---	---
27	CPB	Compliance of the fuel assembly mechanical response to seismic and LOCA forces to the requirements of NUREG-0609. (4.2.3.3)	490.3	4.2.3.5.2	14
28	CPB	A monitoring system needed to detect online fuel rod failure (4.2.4.2)	490.4	Figure 9.3.6-1	14
29	CPB	A description of plans for post irradiation poolside surveillance of fuel. (4.2.4.3)	490.5	4.2.4.6	5
30	CPB	To supply a report to show compliance of the loose parts detection system with RG 1.133. (4.4.8)	492.4	4.4.6.4	14
31	CPB	Itemized documentation required by Item 11.F.2 of NUREG-0737. (4.4.8)	492.5	4.4.6.6, 4.4.6.6.1	14
32	CPB	The use of irradiation strengthened Zircaloy yield strengths must be resolved. (4.2.1.1 and 4.2.3.1)	---	4.2.1.1, 4.2 References	14
33	CPB	The predicted cladding collapse time exceeds the expected lifetime of the fuel. (4.2.3.2)	---	4.2.1.3	14
34	MTEB	If SA-540, Class 1 or 2 materials were used in the reactor coolant pumps, the applicant must demonstrate the adequacy of their fracture toughness. (5.3.1)	---	Table 5.2.3-1	17
35	MTEB	There is no qualification that demonstrates compliance with fracture toughness test requirements. (5.3.1.2)	---	5.3.1.2	17
36	MTEB	Qualification of individuals performing fracture toughness tests needed. (5.3.1.2)	---	5.3.1.2.h)	17
37	MTEB	There must be indication that all RCPB materials not be deleterious to the fracture toughness requirement. (5.3.1.2)	---	---	--
38	MTEB	Meet requirements of GDC 51 for reactor containment pressure boundary. (6.2.7)	---	3.1.4.4	17
39	MTEB	Fracture toughness data needed to justify an exemption from the requirements of NB-2300. (5.3.1.2)	---	5.3.1.5	17
40	MTEB	Revision of the FSAR to indicate the applicability of Westinghouse Topical Report WCAP-9292. (5.3.1.2)	---	---	--

DSER Open Item No.	NRC Branch	Subject (DSER Section)	Safety Review Question	Affected FSAR Section(s)	Amendment
41	MTEB	Provisions of the pressure-temperature limit curve for Unit 2 are not provided. (5.3.2)	---	---	--
42	MTEB	Sufficient information needed to define the capsule withdrawal sequence with Paragraph IV A.1, Appendix G. (5.3.1.3)	---	5.3.1.6	17
43	MTEB	Compliance with reference temperature for all ferretic RCPB materials. (5.3.1.3)	---	5.3.1.5	17
44	MTEB	Identify the location of each material surveillance capsule and the materials in each capsule. (5.3.1.3)	---	5.3.1.6, Figure 5.3.1 3	17
45	RSB	A relief system needed for water trapped between the RHR suction lines from the RCS hot legs. (5.4.7.3)	440.25	5.4.7.2.1	14
46	RSB	A comparison of the Shearon Harris parameters to those listed in WCAP 7769. (5.2.2.1)	440.139	5.2.2.1	14
47	RSB	Further information needed on the reactor coolant vent system. (15.9)	440.142	15.4.12.5	14
48	RSB	Needed details of the use of PORVs to limit pressure to less than scram setpoint with a 50% load reduction. (5.2.2.1)	---	TMI Appendix II.K.3.10	14
49	RSB	Requirements of BTP 5-1, and suitable plant systems to place this plant in a cold shutdown condition. (15.4.7.5)	440.114	5.4.7.2.8	14
50	RSB	To identify the potential interlock failure that could cause more than one valve or component to fail. (5.4.7.2)	440.117	5.4.7.2.6, 8.3.1.1	14, 19
51	RSB	List of components required for operation and support of the ECCS. (6.3.2)	440.38	6.3.2.2	14
52	RSB	Information needed on ECCS interlocks. (6.3.2)	440.119	---	--
53	RSB	To show sufficient time to begin safety injection manually in the case of a small LOCA. (6.3.5.2)	440.120	6.3.2.8	14
54	AEB	Control room habitability remains an open issue because of the ECCS leakage outside containment. (6.3.5.2)	450.5	6.4.1, 6.4.2, 6.4.3, 6.4.4, 6.4.6	14

DSER Open Item No.	NRC Branch	Subject (DSER Section)	Safety Review Question	Affected FSAR Section(s)	Amendment
55	AEB	Spray system incorporates a 100 second delay for the option of additional sodium hydroxide. (6.5.2)	---	6.2.2.2.2.2, 6.2.2.3.2, 6.2.2.5.2, 7.3.1.3.1.1, Table 7.5.1-1	14, 9
56	RSB	Spurious movement of a valve whose mispositioning could cause degradation of the ECCS. (6.3.2)	440.121	---	---
57	RSB	Operator must maintain adequate core cooling following an assumed LOCA. (6.3.1)	440.135	---	---
58	CSB	Additional information concerning the applicant's calculation of the limiting external differential pressure caused by inadvertent containment spray system actuation is required. (6.2.1.1)	480.11	6.2.1.1.3.4	8 & 14
59	CSB	Acceptance of the design-basis break for the reactor cavity subcompartments. (6.2.1.2)	---	6.2.1.2.3	14
60	CSB	Model acceptable, but there is a question regarding a removable neutron shielding in the reactor capacity. (6.2.1.2)	480.13	---	---
61	CSB	Justification needed as to the generic WCAP-9219 results for Shearon Harris. (6.2.1.2)	480.14	6.2.1.2.3, Table 6.2.1-20B, Figures 6.2.1-20B, -21B, -307 & -308	8 & 14
62	CSB	Provisions to be made to provide mass and energy release rate data for the LOCAs. (6.2.1.3)	480.21	---	---
63	CSB	Minimum containment pressure analysis for ECCS performance not acceptable. (6.2.1.5)	480.25 & 26	6.2.1.5, 6.2.1.5.4,	
64	CSB	The containment pressure instrument lines with safety features are classified Safety Class 2. (6.2.4)	480.34	6.2.4.2.4.4	14

DSER Open Item No.	NRC Branch	Subject (DSER Section)	Safety Review Question	Affected FSAR Section(s)	Amendment
65	CSB	Insufficient information on the containment pressure instrument lines associated with the vacuum relief system. (6.2.4)	---	6.2.1.1.3.4, 6.2.4.1, 6.2.4.2.2, 6.2.4.2.4.3, 6.2.4.2.4.4, 7.5.1.8, 6.2.4.2.4.3, Tables 6.2.4-1, 6.2.4-2, and 7.5.1-10, Figures 1.1.1-1 and 1.1.1-2	14, 19
66	CSB	Compliance with BTP CSB 6-4 "Containment Purging During Normal Plant Operation." (6.2.4)	---	3.9.3.2	14
67	CSB	Leakage integrity tests must be conducted periodically on the purge exhaust system. (6.2.4)	480.22	6.2.1.1.2, 6.2.1.5.2	14
68	CSB	A hydrogen recombiner failure alarm should be provided in the main control room. (6.2.5)	480.43	6.2.5.2.1, 6.2.5.2.3, 7.3.1.4.1	14
69	CSB	A description of the actual post-accident hydrogen monitoring system to be installed so that conformance with the requirements of NUREG-0737 Item II.F.I, Attachment 6 can be confirmed. (6.2.5)	480.46	6.2.5.1.1, 6.2.5.1.3, 6.2.5.2.3, Figure 6.2.5-7	14
70	CSB	Location of the outside containment control panels for the hydrogen recombiners. (6.2.5)	480.45	6.2.2.2.1.1, 6.2.2.2.1.2.1, 6.2.5.1.4, 6.2.5.2.4, 6.2.5.2.4, 6.2.5.3.3	14
71	CSB	Type C test required for valves associated with penetrations numbered 8, 12, 15, 16, 35, 36, and 39. (6.2.6)	480.53 & 54	Table 6.2.4-1	14
72	CSB	Isolation valves to be closed during power operation, startup, hot standby, and hot shutdown and every 31 days in accordance with NUREG 0737 Item II.E.4.2. (6.2.4)	480.40, 37	Future	Future

DSER Open Item No.	NRC Branch	Subject (DSER Section)	Safety Review Question	Affected FSAR Section(s)	Amendment
73	RSB	Review and modify procedures for removing and restoring ESF from service. (15.9.3)	---	TMI Appendix, II.K.1.10	14
74	RSB	A conceptual design and evaluation for the anticipatory trip of the Shearon Harris Plant. (15.9.10)	---	TMI Appendix, II.K.3.10	14
75	ICSB	Trip Setpoint and Margins (7.2.2.2)	ICSB-5	7.2.2.2	17
76	ICSB	Response time testing to be done at specified intervals. (7.2.2.3)	ICSB-10	7.2.3, 7.3.1.6.1	17
77	ICSB	Independent verification of the operability of reactor trip breaker shunt and undervoltage coils. (7.2.2.4)	ICSB-22	7.2.2.2.3.10	17
78	ICSB	Re-evaluate the turbine trip circuitry and logic. (7.2.2.5)	ICSB-21	8.3.1.2.30.b).8)	17
79	ICSB	All inputs to the reactor trip system must conform to the requirements of IEEE 279-1971. (7.2.2.6)	---	7.2.1.1.2.e	9
80	ICSB	The Harris design does not have system level ESF manual initiation capability. (7.3.3.2)	ICSB-57	7.3.1.3.2, 7.3.1.3.3., 7.3.1.3.4, 7.3.1.5.7, Table 7.3.1-3	17
81	ICSB	Suitable test jacks should be provided to facilitate testing of the P-4 interlocks. (7.3.3.3)	ICSB-54	---	---
82	ICSB	Review all safety stems to determine if any safety equipment would change state after reset. (7.3.3.4)	ICSB-17 420.6	14.2.12.1.59	17
83	ICSB	Evaluate the effects of high temperatures after high energy line breaks through the IE Bulletin 79 21. (7.3.3.5)	420.9	7.2.2.3.5	17
84	ICSB	The design of feedwater isolation does not meet Paragraph 4.7 of IEEE 279 on "Control and Protection System Interaction." (7.3.3.6)	ICSB-29	7.2.2.3.5	17 (Figures will be sent later)

DSER Open Item No.	NRC Branch	Subject (DSER Section)	Safety Review Question	Affected FSAR Section(s)	Amendment
85	ICSB	Concerns the adequacy of the instrumentation for terminating sodium hydroxide addition and tank isolation valves. (7.3.3.7)	ICSB-30	6.5.2.2, 6.5.2.5, 6.2.2.2.2.2, 6.2.2.3.2 Table 7.5.1-1	9
86	ICSB	A list of non-Class IE control signals that are used as inputs to Class IE control circuits. (7.3.3.8)	ICSB-8	7.2.1.1.8	17
87	ICSB	Ensure that the power supply and control of service water are from the same power source as the CCW pump it replaced. (7.3.3.9)	ICSB-42	---	---
88	ICSB	No description of how the electrical power supply and control is obtained when the spare pump is used. (7.3.3.10)	---	8.3.1.1.2.4	9
89	ICSB	Justification needed for the adequacy of the test program for isolation devices. (7.3.3.11)	ICSB-7	7.7.2.1	17
90	ICSB	Need to address the problem of the removal of the continuity light from the circuit during the testing of the master relays. (7.3.3.12)	---	---	---
91	ICSB	The description of the ESFAS analysis, which is provided in FSAR Section 7.3.2.1, is incomplete. (7.3.3.13)	---	7.3.2.1	11
92	ICSB	Auxiliary feedwater control design incompleted. (7.4.2.1)	ICSB-23, ICSB-2(b)	7.3.1.3.3, 10.4.9, Tables 7.1.1-1, 7.4.1-2 & 10.4.9-2, Figures 7.3.1-9, 7.3.1-9a, 7.3.1-9b, and 7.3.1-10	9
93	ICSB	Capability of the PORV questionable in achieving total shutdown. (7.4.2.2)	ICSB-37	7.4.1.7	17
94	ICSB	Review the adequacy of emergency operating procedures on loss of any Class IE or non-Class IE buses supplying power. (7.5.2.1)	420.5	8.3.1.1.1.4, Table 8.3.1-9	17
95	ICSB	Confirmation needed that valve position indicating switch and reed switch will be qualified. (7.5.2.2)	ICSB-2(a)	TMI II.D.3	9

DSER Open Item No.	NRC Branch	Subject (DSER Section)	Safety Review Question	Affected FSAR Section(s)	Amendment
96	ICSB	Modifications to the LTOP to correct single failure potential. (7.6.2.2)	ICSB-50	7.6.1.11	17
97	ICSB	Valve position and alarm do not satisfy the single failure criterion. (7.6.2.3)	ICSB-49	6.3.5.5.1, 6.3.2.2.9.6	6
98	ICSB	To identify and demonstrate failure of any power source or sensors that provide power or signals to 2 or more control functions will not result in consequences more severe than those of Chapter 15. (7.7.2.1)	420.8	7.7.2.1	Future
99	ICSB	A review requested to determine whether the harsh environment associated with high energy line breaks might cause control system malfunction and result in consequences more severe than those of FSAR Chapter 15. (7.7.2.2)	420.7	Refer to DSER Open Item #98	Future
100	ICSB	A test to be conducted during startup testing to confirm the capability for remote shutdown. (7.4.2.3)	ICSB-33	---	---
101	ICSB	A description of the environmental control system to ensure instrumentation lines are protected from freezing during extremely cold weather. (7.7.1)	ICSB-6	7.7.1.11	9
102	PSB	Provide a detailed discussion of the training proposed for operators, maintenance, QA, and supervision related to emergency diesel generator.	430.12	8.3.1.1.1.5	17
103	PSB	Optimum emergency diesel generator readiness in a nuclear power plant is necessary.	430.13	1.8 (RG 1.108)	17
104	---	Deleted.	---	---	---
105	PSB	In obtaining a safe cold plant shutdown, all working stations must be identified for plant personnel to communicate with the control room or auxiliary control room.	430.15	9.5.2.2.1	17
106	PSB	Provide emergency lighting in areas to ensure safe shutdown and personnel movements.	430.17 and 18	9.5.3.3, Table 9.5.3-1, Figure 9.5.3-1 and -2	17
107	PSB	Provide additional information to demonstrate the integrity of fuel oil piping from storage tanks to the diesel generators under postulated accident conditions.	430.21	9.5.4.3	17

DSER Open Item No.	NRC Branch	Subject (DSER Section)	Safety Review Question	Affected FSAR Section(s)	Amendment
108	PSB	Provide sufficient detail that allows identification of the type of protection provided for the fuel oil storage tank vent and fill lines against tornado missiles.	430.22	9.5.4.3	17
109	PSB	Discuss possible fire hazard of the fuel oil that would be discharged to the roof of the Diesel Fuel Oil Storage Tank Building following a relief valve lift.	430.23	9.5.4.2	17
110	PSB	Discuss detection, preventing, and cleaning of algae in the diesel fuel oil storage tank.	430.24	9.5.4.3	17
111	PSB	Identify the sources where diesel quality fuel oil will be available, distances required to be traveled from the source(s) to the plant, and delivery onside under extremely unfavorable environmental conditions.	430.26	9.5.4.6	17
112	PSB	Provide details and electrical schematics of the control system design which prevents the pumps from running continuously.	430.27	9.5.4.1, 9.5.4.2, 9.5.4.3, 9.5.4.4, 9.5.5.3	17 (Figures will be sent later)
113	PSB	Precautionary measures needed to assure the quality and reliability of the fuel oil supply for emergency diesel generator operation.	430.28	1.8, 9.5.4.5	8
114	PSB	Provide clear indication of the class breaks of the piping between the storage tanks and the diesel generator building and what are the consequences of a fill-piping failure regarding leakage.	430.31 and 32	9.5.4.3, Figure 9.5.4-2	17
115	---	Deleted.	---	---	---
116	PSB	Give details for provisions for minimizing the turbulence in refueling operations as described in ANSI N 195.	430.34	9.5.4.1, 9.5.4.3	17
117	PSB	Additional information needed on the consideration of cooling water leakage with respect to NRC Question 430.35 response.	430.35	9.5.5.3	17
118	PSB	More discussion required than provided in the NRC Question 430.36 response regarding component safety class.	430.36	9.5.4.2, 9.5.4.3, Figure 9.5.4-2	17

DSER Open Item No.	NRC Branch	Subject (DSER Section)	Safety Review Question	Affected FSAR Section(s)	Amendment
119	PSB	Diesel generators are required to start automatically on the loss of power and in the event of a LOCA and capable of operation at less than full load for extended periods without degradation of performance or reliability.	430.42	8.3.1	17
120	PSB	Identify piping supply interface and all safety class breaks regarding diesel generators.	430.47 and 49	9.5.5.3, Figure 9.5.5-1	17
121	PSB	Expand FSAR Section 9.5.6.4 to include a discussion on testing and calibration of the air starting system controls, alarms, indicators, and safety devices.	430.50	9.5.4.5, 9.5.5.4, 9.5.6.4, 9.5.7.4, 9.5.8.4	17
122	PSB	Seismic capability of pressure switch and connecting tubing.	430.51	9.5.6.3, Table 3.2.1-1, Figure 9.5.6-1	17
123	PSB	Describe provisions made in the lube oil system during extended operations.	430.57	9.5.7.3	17
124	PSB	Provide clarification of the industry standards used in instances where the Diesel Generator lubrication system is not designated as ASME Section III Class 3.	430.58	9.5.7.2	17
125	PSB	Clarify the industry standards used in instances where the Diesel Generator Air Intake and Exhaust System is not designated as ASME III Class 3.	430.63	9.5.8.3	17
126	PSB	If the vent line is non-safety, it should be at least seismically supported.	430.64	9.5.8.3	17
127	PSB	Provide detail information on how the turbine speed control and overspeed protection system functions.	430.65	10.2.2	17
128	ASB	Concerns drainage of the New Fuel Pool and conformance of K_{eff} to Standard Review Plan 9.1.1. (9.1.1)	---	9.1.1.1, 9.1.2.1, 9.1.2.3	7
129	ASB	Design the new fuel storage facility which will maintain K effective equal to or less than 0.95 when considering flooding with non-borated water fire extinguishing aerosols.	---	9.1.1.3	14

DSER Open Item No.	NRC Branch	Subject (DSER Section)	Safety Review Question	Affected FSAR Section(s)	Amendment
130	ASB	Explain how draining of the new fuel pool is prevented when spent fuel is being stored. (9.1.1)	---	9.1.1.2, 9.1.3.3	14
131	ASB	Provide information to show how spent fuel will be kept cool in the event of a single failure when only Unit 1 is complete and only one pump and one heat exchanger is available to cool the fuel pools, as noted in Section 1.2.3 of the FSAR. (9.1.3)	---	1.2.3, 9.1.3.3	14
132	ASB	Confirm that the fuel pool heat load estimate for the final full-core off-load given in FSAR Table 9.1.3-1 was obtained using Branch Technical Position ASB 9-2. (9.1.3)	410.8	9.1.1, 9.1.2, 9.1.3 Tables 9.1.3-1 & 9 Figures 9.1.1-1 & 2	7 & 14
133	ASB	Provide a crane load drop analysis as required by NUREG 0612, "Control of Heavy Loads at Nuclear Power Plants". (9.1.5)	410.11(3)	9.1.4.2.2.7, 9.1.4.2.2.8 App. 9.1A	14
134	ASB	Additional information needed on how the removal barrier placed between the cash loading pool and the new fuel pool will prevent the spent fuel cask-handling crane from being inadvertently moved over the new fuel pool. (9.1.5)	410.11(1)	9.1.4.2.2.7, Figure 9.1.4.7	14
135	ASB	Describe precautions of preventing the cask handling crane from approaching within 15 ft. of the new fuel pool. (9.1.5)	410.11(2)	9.1.4.2.2.7, Figure 9.1.4.7	14
136	ASB	Provide more information to ensure that the reactor coolant pump bearing damage due to inadequate cooling and subsequent fuel damage can be prevented. (9.2.2)	410.13	9.2.2.3, 9.2.2.5	14
137	ASB	Identify which heat loads cooled by the CCW are safety related and which are not. (9.2.2)	---	9.2.2.2.1; Table 9.2.2-3	14
138	ASB	Requirements are not met for the testing the quality of air periodically. (9.3.1)	---	Future	Future
139	ASB	Show that the radiation monitors located around the walls of the fuel-handling building are effective in initiating the emergency exhaust subsystem in the event of a fuel-handling accident. (9.4.2)	---	12.3.4.1.8.3	14
140	ASB	Modifications needed for the diesel generator room air intakes with respect to air particulate ingress. (9.4.5)	---	9.4.5.2.5.1; Table 9.4.5-6 Figure 9.4.5-2	10 & 14

DSER Open Item No.	NRC Branch	Subject (DSER Section)	Safety Review Question	Affected FSAR Section(s)	Amendment
141	ASB	No further action taken.			---
142	ASB	No further action taken.			---
143	ASB	No further action taken.	---	---	---
144	CHEB	Provide a procedure for relating radionuclides to estimated degrees of reactor core damage. (9.3.2.2)	---	---	---
145	CHEB	Provide plant procedures for chemical analyses after an accident. (9.3.2.2)	---	---	---
146	CHEB	Provide the schematic diagrams of the sample lines and sampling panels. (9.3.2.2)	---	9.3.2.2.3, Figure 9.3.2.3	14
147	CHEB	Provide information on frequency and type of testing to ensure long-term operability of the Post-Accident Sampling System (9.3.2.2)	---	9.2.3.4	14
148	CHEB	Describe operator training requirements for Post-Accident Sampling System. (9.3.2.2)	---	---	---
149	PSB	Provide a discussion of the inservice inspection program for the extraction steam valves.	430.69	10.2.3.6	17
150	PSB	Explain the protection of the turbine overspeed control system in case of high or moderate energy line failure.	430.70	10.2.4	17
151	PSB	Provide a description of the design of the hydrogen system.	430.72	9.3.9, 9.3.4.1.2.2	17
152	PSB	Provide a consistent design description with the referenced figures for Section 10.4.1.5.	430.78	10.4.1.5	17
153	PSB	Information needed on field inspection frequency and extent of inservice testing and inspection of the Turbine Bypass System.	430.85	10.4.4.4	17
154	PSB	Ability of the turbine overspeed protection system to withstand failure of the turbine bypass.	430.86	10.2.4	17

DSER Open Item No.	NRC Branch	Subject (DSER Section)	Safety Review Question	Affected FSAR Section(s)	Amendment
155	METB	The release of the offgas during hogging operations must be monitored as noted in Table 1 of SRP 11.5. (10.4.2)	460.5	10.4.2.2, 10.4.3.3, 11.5.1.2.2, 11.5.2.5.11, 11.5.2.7.2.9, 11.5.2.7.2.18, Table 11.5.2-2	Future
156	METB	The capacity for MCES is not consistent with guidelines of the "Standards for Steam Surface Condensers". (10.4.2)	460.6	Figure 9.4.4-1 10.4.1.1, 10.4.2.2	10 and 7
157	METB	Describe how the quality group classification, which is non-nuclear safety, correlate with Quality Group D of Regulatory Guide 1.26. (10.4.2)	460.2	9.4.4.2.4, 10.4.2.2	7
158	METB	Correlation of quality group classification to which turbine gland seal system is designed with Quality Group D of RG 1.26. (10.4.3.2)	460.3	10.4.3.1, 10.4.3.2	17
159	METB	The turbine gland seal condenser requires monitoring and sampling according to GDC 64 and Table 1 of SRP 11.5. (10.4.3.2)	460.2	Refer to Open Item #155	17
160	METB	Need to address the Volume Reduction System. (11.1.2)	460.30	11.4.2.1.2, Tables 11.4.1-6, 11.4.1-7, 11.4.1-8	Future
161	METB	Dual gas analyzers required between the compressor and the gas decay tanks per SRP 11. (11.3.2)	---	11.3.2.2.2	17 (Figures will be sent later)
162	METB	No indication of rupture discs in GWPS and liquid seals downstream of the rupture discs. (11.3.2)	460.21	11.3.2.1.4	5
163	METB	Insufficient detail on the VR system. (11.4.2)	460.30	Refer to Open Item #160	Future
164	METB	Drawing (CAR-2165-G-827) that depicts the resin sluice header and the waste concentrate tank. (11.4.2)	460.23	11.4.2.1.1, Figure 11.4.2-6	17

DSER Open Item No.	NRC Branch	Subject (DSER Section)	Safety Review Question	Affected FSAR Section(s)	Amendment
165	METB	Verification of the presence of free water in the waste container and a commitment to re-process the container when free water is discovered. (11.4.2)	---	---	---
166	METB	Description of the dry salt system. (11.4.2)	460.24	11.4.2.1.7	Future
167	METB	Describe the handling of the filter sludge. (11.4.2)	460.27	Future	Future
168	RAB	Provide dose rate maps that identify vital areas. (12.2)	471.21	---	---
169	RAB	Shielding for protection from neutron and gamma-ray streaming and the biological shield into occupiable areas inside containment has not been adequately analyzed. (12.3.2)	471.2	12.3.2.15, Table 12.3.2-9	14
170	RAB	Provide drawing and location of area monitoring instrumentation. (12.3.4.1)	---	11.5.2.7.2.17	14
171	RAB	Commit to place ventilation monitors upstream of relevant ventilation streams or demonstrate alternatives that will ensure that the continuous monitors are capable of detecting 10 mpc hours. (12.3.4.2)	---	12.3.4.2.3	14
172	RAB	To fill staff positions the Radiation Protection Manager qualifications specified in RG 1.8 must be met. (12.5.1)	---	12.5.1.4	14
173	RAB	Environmental and Radiation Control Manager must have access to the General Plant Manager but also be independent of the plant operations management. (12.5.1)	---	Future	Future
174	METB	Details of the vent exhaust from the spent resin storage tanks and decanting tanks. (11.4.2)	460.32(3)	11.4.2	17
175	METB	Heat tracing should be included from the concentrates line of the recycle evaporator to the boric acid tank. (11.4.2)	460.32(1)	9.3.4.2.2	17
176	METB	Provide the Process Control Program (PCP). (11.4.2)	---	11.4.1.2, 11.4.2	17, 19
177	METB	Monitoring and sampling of the turbine gland seal condenser exhaust and the mechanical vacuum pump is not necessary because expected radiation levels fall below guideline values. (11.5.2)	460.38	Refer to Open Items #155 and 181	17

DSER Open Item No.	NRC Branch	Subject (DSER Section)	Safety Review Question	Affected FSAR Section(s)	Amendment
178	METB	The condenser vacuum pump effluent is not sampled in accordance with Table 1. (11.5.2)	460.38	Refer to Open Items #155 and 181	17
179	METB	Sample provisions and an effluent monitor for the turbine building vents are not provided. (11.5.2)	---	9.4.0 and Table 9.4.0-2 11.5.1.3.1 (Refer to Open Items #155 and 180)	15 and 17
180	METB	Both containment pre-entry purge and the continuous containment purge shall be monitored continuously. On a high radiation signal isolate purge operation in an automatic fashion. (11.5.2)	460.48	11.5.2.7.2.15 12.3.4.2.8.1, 12.5.2.7.2.15	5 Future
181	METB	Continuous sample of the service water system required by Table 2. (11.5.2)	460.36	11.5.2.7.2.1, Table 11.5.5-2 (Refer to Open Item #155)	17 (Figures to be sent later)
182	METB	Nationally Recognized Standard should be NBS Traceable as related to position C of RG 4.15. (11.5.2)	460.42	11.5.2.1.1	17
183	METB	Address information required by II.F.1, Attachment 1. (11.5.2)	471.22	Future	Future
184	METB	Address information required by II.F.1, Attachment 2. (11.5.2)	471.22	Refer to Open Item #230	17
185	METB	Pumps not complying with criteria II.2.a of SRP 1.5. (11.5.2)	460.41X	11.5.1.3.1, Table 11.5.1-1, Figure 11.2.2-6 11.5.1.3.1, Table 11.5.1-1	5 and 17
186	METB	There is no capability to replace or decontaminate monitors used for gaseous effluents without opening the process system. (11.5.2)	460.43	11.5.2.1.1	5 and 17
187	METB	The diversion valves associated with the turbine building drain monitor do not fail in the closed position. (11.5.7)	460.44	11.5.2.7.2.5	17

DSER Open Item No.	NRC Branch	Subject (DSER Section)	Safety Review Question	Affected FSAR Section(s)	Amendment
188	METB	Conformance of non-ESF instrumentation to the design guidance of Appendix 11.5-A of SRP 11.5 and control of radioactive liquids has not been addressed. (11.5.2)	460.45	11.5.1.1, 11.5.1.2.1	5 and 17
189	METB	Capability to determine the quantity of effluents released from turbine building drain monitors and the tank area drain monitors. (11.5.2)	460.47	Refer to Open Item #187	17
190	METB	Determine whether releases from the vents of the boron recycle system are routed so that adequate monitoring and sampling of these sources is present. (11.5.2)	---	11.3.2, 9.3.4.2.2, Figures 9.3.4-6, 9.3.4-7	17
191	METB	Contradictory information presented on the operation of the vent valve on the WPB cooling water surge tank. (11.5.2)	460.35	11.5.2.7.1.5	5
192	METB	The review of the process and effluent monitoring system cannot be complete because some items have not been addressed. (11.5.2)	---	Future	Future
193	METB	Integrity of Systems Outside Containment likely to contain radioactive material for PWRs and BWRs.	---	TMI Appendix III.D.1.1	17
194	PSRB	Review by NSSS vendor of all preoperational and startup test procedures, and the emergency operating procedures. (13.5.2.3)	---	---	---
195	PSRB	Required to develop emergency operating procedures based on the revised Westinghouse Owners' Group Emergency Response Guidelines. (15.8)	---	---	---
196	SGPR	Meet requirements of the Industrial Security Plan, Safeguards Contingency Plan, and Security Personnel Training and Qualification Plan. (15.5.2.3)	---	---	---
197	EPB	Review Emergency Plan when available. (13.3)	---	---	---
198	PSRB	To provide hot containment penetration cooling acceptance criteria at a later date. (14)	640.5.5.w	14.2.12.1.28	7
199	PSRB	To provide appropriate acceptance criteria for the radiation monitoring system test 14.2.12.1.14. (14)	640.20(2)	14.2.12.1.14	7
200	PSRB	Revise current proposed method of measuring process-to-sensor response time. (14)	640.28	14.2.12.1.11, 15.0.6, 14.2.12.2.19	---

DSER Open Item No.	NRC Branch	Subject (DSER Section)	Safety Review Question	Affected FSAR Section(s)	Amendment
201	PSRB	To provide flow data for relief valve capacity acceptance criteria at a later date. (14)	640.32	15.1.4.2	14
202	PSRB	Ensure that a special lower power testing program is conducted according to Enclosure 9. (14)	640.5.4.+	13.2.1.2, 14.2.12.1.81, 14.2.12.2.26, 14.2.12.1.25, 14.2.12.1.79, 14.2.12.1.16 14.2.12.1.17	8, 11, & 14
203	PSRB	Section 14.2.7 must be revised to commit to the applicable revisions of RG 1.95 and RG 1.140. (14)	640.12	14.2.7	7
204	PSRB	Containment recirculation fan testing must be incorporated into the actual test abstract. (14)	640.5.1.h(9)	14.2.12.1.57	7
205	PSRB	Sump level control must be modified to indicate under which specific test abstract the operation of each level control switch will be demonstrated. (14)	640.5.1.j(20)	14.2.12.1.40	---
206	PSRB	Correlation between instruments listed in FSAR Section 7.5 and the applicable test abstracts. (14)	640.5.1.j(22)	7.5.1.9	14
207	PSRB	Post-accident ventilation system performance must be incorporated into the actual test abstract. (14)	640.22	14.2.12.1.58	7
208	PSRB	Conduct a 24 hour endurance test on all AFW system pumps and demonstrate the pumps remain within design limits. (14)	640.5.4.k	14.2.12.1.34, 14.2.12.2.32	14
209	RSB	Supply a list of assumed single failures utilized in the FSAR Chapter 15 analysis. (15)	440.123	15.0.13	14
210	RSB	Analyze the CVCS malfunction event in combination with the most limiting single failure and confirm acceptance criteria is met. (15.5)	440.132	15.5.1.2.e) Table 15.6.5-6	14
211	AEB	Evaluate radiological consequences with respect to the maximum operational ECCS leakage outside containment. (15.6.5.2)	450.5	15.6.5.4.2 Table 15.6.5-6	14
212	RSB	Reanalyze the reactor coolant pump rotor seizure and shaft break using stated criteria. (15.3.3 and 15.3.4)	440.83	15.3.3.4, 15.3.4	14

DSER Open Item No.	NRC Branch	Subject (DSER Section)	Safety Review Question	Affected FSAR Section(s)	Amendment
213	RSB	The requirements of SRP 15.4.6 are not met. (15.4.6)	440.86	15.4.6.2.c), 15.4.6.3.b) Table 15.4.6 1	14
214	RSB	Additional information needed on the adequacy of the low temperature overpressure protection system. (5.2.2.2)	440.18, 106 and 140	5.2.2.11.1, 5.2.2.11.2	14
215	RSB	Analyze consequences of intermediate streamline breaks from full power, with the offsite power being lost at the worst time and reactor trip occurring on the first safety-grade trip. (15.1.5)	440.109	15.1.5.1	14
216	RSB	The stream generator tube rupture accident did not support the isolation time of 30 minutes. (15.6.3)	440.111, 133 and 134	Future	Future
217	PSRB	Develop guidelines for reactivity control by Westinghouse Owners' Group with respect to mitigation of ATWS events. (15.8)	---	---	---
218	---	Deleted.	---	---	---
219	RSB	Review the radiological consequences of a steam generator tube rupture accident. (15.6.3)	450.04	Future	Future
220	QAB	Evaluate structures, systems, and components that are under control of the QA program. (17.5)	260.67	5.2.1.2, Tables 3.2.1-1 & 5.2.1-2	14
221	HFEB	Proposed schedule of addressing all points as required by SECY 82-111 and DCRDR. (18)	---	---	---
222	---	Deleted.	---	---	---
223	ICSB	Use SECY-82-111 (Supplement 1 of NUREG-0737) in the implementation of emergency response capability. (7.5.2.2)	ICSB-44	1.8 (RG 1.97)	Future
224	ASB	Circulating water system does not meet the requirements of GDC 4; and, therefore, unacceptable. (10.4.5)	---	10.4.5.3	14
225(1)	ASB	There has been no response to the letter concerning the AFWS design. (10.4.9)	410.22(1)	Table 10.4.9-3 Section 10.4.9.3	14

DSER Open Item No.	NRC Branch	Subject (DSER Section)	Safety Review Question	Affected FSAR Section(s)	Amendment
225(2)	ASB	Response should include reliability evaluation as discussed in NUREG-0611. (10.4.9)	410.22(2)	Appendix 10.4.9.A	14
225(3)	ASB	Response should include point by point review of the AFW system design, Technical Specifications, and operating procedures. (10.4.9)	410.22(3)	9.2.6.5, 10.4.9.3	14
225(4)	ASB	A discussion of the design basis for AFWS flow requirements and verification that AFWS will meet these requirements. (10.4.9)	410.22(4)	10.4.9.1, 15.2.6, 10.4.9.13	14
225(5)	ASB	Discrepancies between the description of valve operators given in FSAR Figure 10.1.0-3 and those given in FSAR Figures 7.3.1 9 and 9.3.1 10. (10.4.9)	410.22(5)	7.3.1.9, 7.3.1.9a, 7.3.1.9b, 7.3.1.10, 10.1.0.3	5, 9, and 14
226	METB	Confirmation needed that chemical drain wastes are sent to the RO concentrates tanks. (11.1.1)	---	11.2.2.2	17
227	---	Deleted.	---	---	---
228	METB	Figures 9.3.2 1 and 10.1.0 6 to show sampling points P-11, 21, and 13 of the steam generator blowdown. (11.5.2)	460.34	9.3.2.2.1, 10.4.8.4, 11.5.2.7.1.3 Table 9.3.2-1	17
229	METB	Table 9.3.2 2 Figure 9.3.2 2, and Section 9.3.2.2.2 to reflect grab sampling of the service water system. (11.5.2)	---	Figures 9.3.2-1, 9.3.2-1a, 10.1.0-6 (Refer to Open Item #181)	17
230	METB	Table 11.5.2 1 to reflect the high range monitoring requirements of NUREG-0737, II.F.1, Attachments 1 and 2. (11.5.2)	---	11.5.2.7.2.16, Tables 12.3.4-1, 11.5.2 2	17
231	METB	Address the monitoring of the effluent from the volume reduction system. (11.5.2)	460.30	11.4.2.1.2	Future
232	---	Deleted.	---	---	---

DSER Open Item No.	NRC Branch	Subject (DSER Section)	Safety Review Question	Affected FSAR Section(s)	Amendment
233	---	Deleted.	---	---	---
234	---	Deleted.	---	---	---
235	RSB	Provide a response to NUREG 0737 Item II.K.3.17, "Report on Outages of Emergency Core Cooling Systems License Report and Proposed Technical Specification Changes."	---	TMI Appendix, II.K.3.17	14
236	AEB	The response time, negative pressure, and the location of the radiation monitors during fuel handling are unresolved. (15.7.4.2)	---	6.5.1.1.1, 6.5.1.2.1, 15.7.4.1	14
237	CPB	Interim position for operating reactors that consists of a restriction on operations above 90% power so that either the reactor is in manual control or rods are required to be out more than 215 steps. (15.4.3)	---	15.4.3.2	14
238	MEB	Conflict with quality group classifications. (3.2.2)	---	5.2.1.2, Tables 3.2.1-1, 3.2.1 2	14
239	---	Deleted.	---	---	---
240	---	Deleted.	---	---	---
241	RSB	Address the provision of alternate source of cooling. (5.4.7.2)	---	---	---
242	CSB	Complied with the provisions of NUREG 0737, Item II.E.4.2, with the exception of the containment isolation setpoint pressure. (6.4.2)	---	6.2.4.2.2	14
243	MEB	Quality group required for the reactor vessel internals. (3.2.1)	210.01	Table 3.2.1-1	9
244	MEB	Explain inconsistency of several waste processing system components that are identified safety class 3, but not Seismic Category I. (3.2.1)	210.02	Table 3.2.1-1	9
245	MEB	Identify safety class 2 systems or components that are part of the reactor coolant pressure boundary. (3.2.1)	210.03	---	---
246	MEB	Provide details of the portions of the safety injection system. (3.6.2)	210.04	---	---

DSER Open Item No.	NRC Branch	Subject (DSER Section)	Safety Review Question	Affected FSAR Section(s)	Amendment
247	MEB	Effects of jet impingement criteria (3.6.2)	210.05	---	---
248	MEB	Specify the assumed damage by an unrestrained whipping pipe to an impacted pipe of equal size with thinner wall thickness. (3.6.2)	210.06	3.6.1.3	9
249	MEB	Class 1 piping break postulation criteria. (3.6.2)	210.07	3.6.2.1.1.2	17
250	MEB	Clarify your positions with respect to branch connections being considered terminal ends. (3.6.2)	210.08	3.6.2.1.1.2	9
251	MEB	Breaks should be postulated at all terminal ends in ASME Class 2 and 3 piping. (3.6.2)	210.09	3.6.2.1.1.3	9
252	MEB	The break exclusion region for the main steam line should only extend to the inboard or outboard isolation valves. (3.6.2)	210.10	3.6.2.1.4, 6.6.8, Figure 3.6.2-1	9
253	MEB	Justify not evaluating pipe whip and jet impingement loads for main stream and feedwater lines in the steam tunnel. (3.6.2)	210.11	3.6.2.1.4, 6.6.8, Figure 3.6.2-1	9
254	MEB	Insufficient detail for a complete review of your dynamic analysis of jet thrust exists. (3.6.2)	210.12	3.6.2.2.1, Figure 3.6A-1 through -27	9
255	MEB	More details of the methods used to perform piping dynamic analysis is required. (3.6.2)	210.13	---	---
256	MEB	Verify targets of unrestrained whipping pipes and jet impingement have been considered. (3.6.2)	210.14	---	---
257	MEB	Justify your jet expansion model for saturated water blowdown or change your FSAR. (3.6.2)	210.15	3.6.2.3.4.2	9
258	MEB	Justify the use of limited area circumferential or longitudinal break, and provide a list of limited break areas. (3.6.2)	210.16	3.6.2.5.1.a)	17
259	MEB	Provide details and examples of the analysis performed with respect to piping restraints. (3.6.2)	210.17	---	---

DSER Open Item No.	NRC Branch	Subject (DSER Section)	Safety Review Question	Affected FSAR Section(s)	Amendment
260	MEB	Dynamic testing of crushable pipe restraints. (3.6.2)	210.18	---	---
261	MEB	For review, provide primary and secondary stress intensity ranges in the main reactor coolant loop fatigue analysis. (3.6.2)	210.19	Table 3.6.2 2	9
262	MEB	Provide a review of the summary data developed to select postulated break locations for balance of plant piping. (3.6.2)	210.20	Tables 3.6A-15, -16, -17.1, -17.2, -17.3, -18, Figure 3.6A-1	9
263	MEB	Closely simulate the dynamic behavior of the subsystems (3.9.2.2)	210.21	---	---
264	MEB	Define the term "significant modes" in seismic subsystem analysis. (3.9.2.2)	210.22	3.7.3	19
265	MEB	Discuss special devices that have been used to eliminate the effects of relative displacements. (3.9.2.2)	210.23	3.7.3.1.1	
266	MEB	Explain where the equivalent static load method is used. (3.9.2.2)	210.24	---	---
267	MEB	Further discussion of modal acceleration and alternate response spectra is required. (3.9.2.2)	210.25	3.7.3.8.1, Figure 3.7.3-1	9
268	MEB	Provide an example of your computer method analysis using the response spectra method of Section 3.7.3.1.1 (3.9.2.2)	210.26	---	---
269	MEB	Stress in component supports due to differential seismic motion are not treated as secondary by ASME Subsection NF. (3.9.2.2)	210.27	See DSER Open Item #275	17
270	MEB	Provide for review the ASME code service limits. (3.9.1)	210.28	Tables 3.9.1-2, 3.9.1-3	9
271	MEB	Specify cases where you have combined loads by algebraic addition. (3.9.1)	210.29	3.9.1.2.2	9
272	MEB	The computer program in analyses of seismic Category 1 code and non-code require additional information. (3.9.1)	210.30	3.9.1.2.2	9
273	MEB	Clarify the statement concerning plastic component analysis. (3.9.1)	210.31	3.9.1.4.7	9

DSER Open Item No.	NRC Branch	Subject (DSER Section)	Safety Review Question	Affected FSAR Section(s)	Amendment
274	MEB	Acceptance criteria for piping vibration testing is unacceptable. (3.9.2.1)	210.32	2.12.1.12, 3.9.2.1.1, 3.9.2.1.3, 14.2.12.1.12	9 and 17
275	MEB	Explain criteria for seismic/non-seismic interface anchors. (3.9.2.2)	210.33	3.7.3.1.1, 3.7.3.9.1, 3.7.3.9.2, 3.7.3.13	9 and Future
276	MEB	Provide a discussion giving a more detailed justification to Regulatory Guide 1.48; Regulatory Positions C.6, C.7, C.8, and C.10. (3.9.2.1)	210.34	1.8 (RG 1.48)	9
277	MEB	Insufficient information on loading combinations, system operating transients, and stress limits. (3.9.3.1)	210.35	3.9.3.1, 3.9.3.1.1, 3.9.3.1.2.2, Table 3.9.3-7	9 and 17
278	MEB	Provide faulted allowable stresses for bolts used in ASME Code components. (3.9.3.1)	210.36	---	---
279	MEB	Provide valve discs limits. (3.9.3.1)	210.37	---	---
280	MEB	The actual stress limits used should be clarified rather than a reference to the appropriate code paragraph. (3.9.3.1)	210.38	Tables 3.9.3-10, 3.9.3-11	9
281	MEB	Define closely spaced modes as used in Responses Spectra Analysis. (3.9.3.1)	210.39	3.9A	9
282	MEB	Identify areas where the piping and support system for pressure relief devices uses hydraulic snubbers. (3.9.3.1)	210.40	3.9.3.4.2	17
283	MEB	Confirm that pressure-relieving devices are spaced according to Regulatory Guide 1.67. (3.9.3.2)	210.41	---	---
284	MEB	Information needed where thermal stress has been considered. (3.9.3.3)	210.42	Table 3.9.3-1	9
285	MEB	Document maintenance records due to problems dealing with inoperable and incorrectly installed snubbers. (3.9.3.3)	210.43	---	---

DSER Open Item No.	NRC Branch	Subject (DSER Section)	Safety Review Question	Affected FSAR Section(s)	Amendment
286	MEB	Provide the program for leak rate testing of those valves that provide an interface between reactor coolant pressure boundary and low pressure systems. (3.9.6)	210.44	3.9.6.2	9 and 11
287	MEB	Provide a program for the inservice testing of pumps and valves including any request for relief from ASME Section XI requirements. (3.9.6)	210.45	Appendix 3.9D	---
288	---	Deleted.	---	---	---
289	EQB	Define critical parameters, design-basis, events, and commitments to IEEE 323 1974 and 344 1975. (3.10.1)	---	3.9.3.2, 3.9.3.2.1.1, 3.9.3.2.1.2	14
290	EQB	Standards in SRP 3.10, IEEE 323 1974 and 344 1975 must be followed. (3.10.2)	---	3.9.3.2.2.1, 3.9.3.2.2.2, 3.10.1.2	14
291	EQB	Submit environmental qualification program for safety-related electrical and mechanical equipment located in harsh environmental areas. (3.11)	---	3.10.1.2	14
292	---	Deleted.	---	---	---
293	MTEB	Insufficient information to demonstrate the generic fracture toughness of ferritic plates. (5.3.3)	---	Table 5.2.3-1	17
294	RSB	Additional information needed concerning the RWST enclosure to ensure the RWST vent line is adequately protected from freezing. (6.3.1)	---	6.2.2.2.3.2	14
295	---	Deleted.	---	---	---
296	---	Deleted.	---	---	---
297	---	Deleted.	---	---	---
298	RSB	LOCA analysis resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary is acceptable. (15.6.5)	---	---	---
299	RSB	Calculated performance of the ECCS following a LOCA accident is considered acceptable.	---	---	---

DSER Open Item No.	NRC Branch	Subject (DSER Section)	Safety Review Question	Affected FSAR Section(s)	Amendment
300	---	Deleted.	---	---	---
301	---	Deleted.	---	---	---
302	---	Deleted.	---	---	---
303	CSB	Revise Table 6.3.4 1 to show the correct General Design Criteria to which the penetrations therein listed are designed.	---	3.1.4.8, 6.2.4.2.4.2, 6.2.4.2.4.4 Tables 6.2.4-1 and 6.2.4-2	14
304	PSB	Provide a discussion of fusing of the DC System to show what happens to maximum short circuit is applied.	430.113	8.3.1.1.2.15.f), 8.3.2.1.3	17
305	PSB	Discuss how non-class IE AC loads are connected to class IE busses in conformance with Regulatory Guide 1.75 for power systems.	430.105	8.3.1.1.2.5, 8.3.1.2.14	17
306	PSB	The response to NRC Question 430.102 does not provide the correct rated load test sequence for the diesel generator load acceptance testing.	430.102	8.3.1.1.2.14.K	17
307	PSB	Describe how the Shearon Harris design meets BTP PSB-1 "Adequacy of Station Electric Distribution System Voltages."	430.93	8.3.1.1.2.1, 8.3.1.1.2.3, 8.3.1.1.2.5, 8.3.1.1.2.11, 8.3.1.1.2.15, 8.3.1.1.3, 14.2.12.82	17
308	PSB	The response to Question 430.97 does not address single failure criteria or include passive accumulator valves.	430.97	6.3.1, 8.3.1.2.3.8	17
309	PSB	Discuss how non-class IE DC loads are connected to class IE DC systems and conformance with Regulatory Guide 1.75 for DC systems.	430.105	1.8 (RG 1.75), 8.3.1.2.14	---
310	PSB	Discuss the emergency diesel generator load sequence reliability.	430.100	Remains an Open Item	

DSER Open Item No.	NRC Branch	Subject (DSER Section)	Safety Review Question	Affected FSAR Section(s)	Amendment
311	PSB	A battery discharge alarm system should be provided with at least an alarm in the control room.	430.106	8.3.2.1.2, 8.3.2.1.3	17
312	CHEB	Report on "Safe Shutdown Analysis in Case of Fire."	---	---	---
313	CPB	Discrepancy found between the power distribution peaking factor, Fq, described in Section 4.3, Nuclear Design and that used in Section 15.6, LOCA Analysis.	---	4.3.2.2.6	14
314	EPB	Respond to NUREG-0737 Supplement 1 requirements for Emergency Response Facilities.	---	13.3	14
315	EPB	Submit local and state emergency plans.	---	13.3	11 & 14
316	LQB	Provide emergency training to Reactor Operators and Senior Reactor Operators.	630.1	13.2	14
317	LQB	Discuss the certifications completed pursuant to Sections 55.10(a)(6), 55.33(a)(4) and (5) of 10 CFR Part 55.	630.2	13.2	14
318	LQB	Provide the title of the individual who is responsible for development, implementation, and administration of licensed and non-licensed operator training.	630.3	13.2	14
319	LQB	For hot license operators and SRO candidates, comply with the requirement of 3 months on shift experience, after 3 months of station operation.	630.4	13.2	14
320	LQB	Provide qualifications of instructors.	630.5	13.2	14
321	LQB	Provide a training program for mitigating core damage as described in II.B.4 of NUREG 0737.	630.6	13.2	14
322	LQB	Additional information requested concerning training/retraining of SHNPP Shift Technical Advisors (STAs).	630.7	13.2	14
323	LQB	Provide detail discussion of the fire brigade training program.	630.8	13.2	14
324	MTEB	Determine if there was a maximum yield strength specified for austenitic stainless steels, and its magnitude.	252.1	4.5.2.1, 5.2.3.4	17

DSER Open Item No.	NRC Branch	Subject (DSER Section)	Safety Review Question	Affected FSAR Section(s)	Amendment
325	MTEB	Specify whether martensitic stainless steels were used in the fabrication of the control rod drive mechanism.	252.2	4.5.1.1, 4.5.2.1, Table 4.5.1-1	17
326	MTEB	Identify the materials used based upon Code Case 1618.	252.3	---	---
327	MTEB	Provide data and rationale as to the acceptability of the electroslag welds in steam generators 1 and 2.	252.4	1.8 (RG 1.34)	17
328	MTEB	Provide rationale and data to demonstrate underclad cracking is not present in carbon-manganese steel pressure vessels.	252.5	5.2.3.1, 5.4.10.2.2	17
329	MTEB	Provide rationale and data as to the acceptability of the fracture toughness of ferritic steel components in the engineered safety features.	252.6	---	---
330	MTEB	Provide rationale why fracture toughness criteria was applied to Class 2 components of the main stream and feedwater systems and not applied to Class 3 components.	252.7	10.3.6.1	17
331	MTEB	Identify materials by specification, grade, type, etc. rather than by proprietary designations.	252.8	Tables 5.2.3-1, 5.2.3 2	---
332	MTEB	Probability of unacceptable damage to safety-related systems and components due to turbine missiles is acceptably low. (3.5.1.3)	---	Future	Future
333	SEB	The model is described erroneously as fixed base.	---	3.7.2.4A	14
334	SEB	The Stardyne code requires further explanation regarding modal damping of the structures and spring representation of the bed rock.	---	3.8.B3	14
335	SEB	Discussion and verification of bed-rock representation as one-way spring by Stardyne when the stability analysis was performed.	---	3.7.4.4A	14
336	SEB	Provide discussion and verification of the concrete crack model in SHELL Code.	---	3.8.B.5	14
337	SEB	Provide discussion of concrete crack model in Stardyne static analysis.	---	3.8.1.4.4.4	14
338	SEB	Justify the use of thin flat plate application to a thick structure member in the Stardyne code.	---	3.8.3.4.1(C)	14

DSER Open Item No.	NRC Branch	Subject (DSER Section)	Safety Review Question	Affected FSAR Section(s)	Amendment
339	SEB	Tabulate the loadings at the anchor point of mainsteam and feedwater pipes on containment wall.	---	---	---
340	SEB	Justification needed for not using dynamic factor on LOCA pressure loading in containment analysis and for not considering an oval mode if shell dynamic response.	---	3.8.1.3.1(h)	14
341	SEB	Explain the difference in acceleration and moment at common points between detailed tank analysis and overall tank-building analysis.	---	3.8.4.1.6	14
342	SEB	Address the changes in the margin of safety when a sloshing effect is included in the fuel handling building earthquake analysis.	---	3.8.4.1.4	14
343	MEB	Specific information needed on the 3 types of supports as related to the design and construction of ASME Class 1, 2, and 3.	---	3.9.3.1, Tables 3.9.3-7, 3.9.3 15, Figure 3.9.3-2	9 and Future
344	PSB	It is not apparent that the design of the on site power system and the switchyard components meets the requirements of GDC-18.	430.90	8.3.1.1.2.4	17
345	PSB	Identify loads which are manually connected to the safety buses and justify the application of these manually connected loads.	430.101	8.3.1.1.2.14.f), Table 8.3.1-2, Figures 8.3.1-12, 8.3.1-13	Future
346	SGEB	Details are needed on the design of the Retaining Wall because of the concern that it might fail.	---	---	---
347	SGEB	Provide information on the methods used to complete lateral earth pressures on building walls.	---	---	---
348	SGEB	Provide foundation properties for soil-structure interaction.	---	---	---
349	SGEB	After inspecting the water control structures, expect the implementation of Regulatory Guide 1.127.		---	---
350	CHEB	Conform the plant fire protection program to the guidelines of BTP CMEB 9.5.1 (NUREG 0800), July 1981 and justify any deviations.	280.1	---	---

DSER Open Item No.	NRC Branch	Subject (DSER Section)	Safety Review Question	Affected FSAR Section(s)	Amendment
351	CHEB	Verify that the closing of the fire doors will be supervised by one of the measures stated in BTP CMEB Section C.5.a.		280.11	9.5.1.2.2
352	CHEB	Self-contained 8 hour minimum capacity, battery powered emergency lighting units be installed in conformance with BTP CMEB 9.5 1 Section C.5.g.	280.24	9.5.1.2.3	14
353	CHEB	Provide an engineered oil containment and collection system for the reactor coolant pump lube oil system to comply with Section D of Appendix R to 10 CFR Part 50.	280.30	---	---
354	MEB	Show where Specification CAR SH M 30, Rev. 16 dated 4 18 83 defines the loading combinations and stress limits for Class 1, 2, and 3 piping.	---	---	---
355	ASB	Show that ESCWS and WPBCWS are protected against internally generated missiles.	---	9.2.8.3	14
356	ASB	Show that the safety-related systems are not adversely affected by a break in the non-essential services chilled water system.	---	3.6.1, 9.2.9.3	14
357	ASB	Detect leakage to the reactor coolant drain tank and pressurizer relief tank.	---	5.2.5.2.1	14
358	ASB	Show that missile fragments resulting from the failure of a rupture disc in the pressurizer relief tank will not adversely affect safety-related equipment.	---	5.4.11.3	14
359	ASB	Give a detailed description of the new fuel racks to enable us to make an independent evaluation of K_{eff} in the new fuel storage pools.	---	9.1.1.1, Table 9.1.1-1	14
360	ASB	Provide detailed information regarding all fuel to be stored, distributed, storage racks and calculation methods used in the determination of K_{eff}.	---	9.1.2 1, Table 9.1.2-1	14
361	ASB	Commit to the installation of both fuel pool cooling pumps and heat exchanger trains in Unit 2 before placing Unit 2 fuel pools cooling system in operation.	---	9.1.3.1, 9.1.3.2, 9.1.3.3, 1.2.3	14
362	ASB	Explain loads equal in weight or lighter and their kinetic energy.	---	9.1.2.3	14
363(1)	ASB	---	---	---	---
363(2)	ASB	Commit to completing Phase I of NUREG 0612 prior to issuance of an operating license.	---	---	---

DSER Open Item No.	NRC Branch	Subject (DSER Section)	Safety Review Question	Affected FSAR Section(s)	Amendment
363(3)	ASB	Commit to completing Phase II of NUREG 0612 prior to the end of the first refueling.	---	---	---
363(4)	ASB	Show that a loss of one phase of a 3 phase electrical system or phase reversal will not result in dropping of a heavy load from any crane intended for such use.	---	9.1.4.3.2	14
364	ASB	Periodic testing of Service Water System	---	---	---
365	ASB	Show that all nonsafety-related heat loads in the Component Cooling Water System are isolated for safety-related loads in the event of suitable emergency initiating signals.	---	9.2.2.1, 9.2.2.2, 9.2.2.2.2, 9.2.2.2.3, 9.2.2.5, 9.1.3.3 Table 9.2.2-3 & -4	14
366(1)	ASB	Show that the ESCWS supplies enough water to the safety-related components to cool them in the event of an accident.	---	9.2.8.3	14
366(2)	ASB	---	---	---	---
366(3)	ASB	Show how the ESCWS has been designed to permit functional and pressure testing.	---	9.2.8.4	14
367	ASB	Show that a break in a NESCWS pipeline as a result of a seismic event will not result in functioning of any safety-related equipment.	---	3.6.1.2.4	14
368	ASB	Show how radioactive fluids are prevented from entering the service water system in the event of a leak into the Waste Processing Building Cooling Water System.	---	9.2.10.2, 9.2.10.2.2	14
369	ASB	Commit to periodic testing of instrument air quality.	---	9.3.1.4	14
370(1)	ASB	Clarify the operation sequences for the CRACS after an emergency signal and justify each sequence on the basis of the operator's safety and maintaining the plan in safe condition.	---	6.4.3, 9.4.1.3	14
370(2)	ASB	Show how the control room area air conditioned system complies with Positions C.7 and C.14 of Regulatory Guide 1.78.	---	2.2.3.3, 9.4.1.2, 9.4.1.2.2	14

DSER Open Item No.	NRC Branch	Subject (DSER Section)	Safety Review Question	Affected FSAR Section(s)	Amendment
370(3)	ASB	Show how the CRACS complies with the guidelines of Position C.4.a and Regulatory Guide 1.95.	---	9.4.1.2.1, 9.4.1.3	14
370(4)	ASB	Show that the CRACS can maintain suitable control room conditions under the most adverse conditions.	---	9.4.1.3	14
370(5)	ASB	Demonstrate that the control room will not be adversely affected by a pipe crack or break on the pressurized area.	---	9.4.1.1	14
371	ASB	Assure that all air intakes to the diesel generator room will comply with recommendations A.2 and C.1 of NUREG/GR 0660.	---	9.4.5.2.5.1, 9.4.5.2.5.4, 9.4.5.2.5.5	14
372	ASB	Show that the failure of the turbine impeller of the steam turbine of the turbine driven SFW pump cause loss of the A&B trains of the motor driven AFW pumps.	---	10.4.9.1, 7.4.1.3, 10.4.9.3	14
373	ASB	Provide assurance that at least 475 gpm will be provided to the steam generators by the AFW system after a loss-of-feedwater (LOFW) event.	---	10.4.9.1, 7.4.1.3, 10.4.9.3	14
374	ASB	Technical Specification with 2 and 3 pumps.	---	---	---
375	ASB	Evaluate the possibility of water hammer because of trapped air at high points.	---	10.4.9.2.1	14
376	ASB	Provide procedures for providing water to the AFW pumps for an emergency startup in the event the CST is unavailable.	---	10.4.9.2.2	14
377	ASB	Provide procedures for valve alignment checks after testing in addition to those done after operating and maintenance.	---	10.4.9.4	14
378	ASB	Technical Specification for flow testing after extended shutdown.	---	---	---
379	ASB	Operator must have 20 minutes following the alarm indicating low level in the CST to switch to a new supply.	---	9.2.6.5, 10.4.9.5	14
380	ASB	The applicant should provide a means for protecting the AFW pumps from the possible failure of the single value in the single supply line from the CST.	---	7.3.1.3.3, 10.4.9.5	14

DSER Open Item No.	NRC Branch	Subject (DSER Section)	Safety Review Question	Affected FSAR Section(s)	Amendment
381	ASB	Verify that a break in the line delivering steam to the turbine of the AFW turbine-driven pump will still permit the AFW system to deliver the necessary coolant flow to the undamaged steam generator.	---	10.4.9.3	14
382	ASB	Show that the moveable barrier between cask pool and the northern fuel pool will not damage spent fuel or safety-related equipment as a result of a seismic event.	---	9.1.4.2.5	14
383	ASB	State whether missiles generated by the failure of fan components in HVAC equipment were considered in the analysis of internally generated missiles.	---	3.5.1.1.2	14
384	EPB	Emergency Planning	---	---	---
385	MTEB	Preservice Inspection Program	---	Future	Future
386	MTEB	Specify whether martensitic stainless steels were used in the fabrication of the control rod drive mechanism.	---	4.5.1.1, 4.5.2.1, Table 4.5.1-1	17
387	PSB	Clarify reference to a permissive signal in Chapter 7 of the FSAR.	---	8.3.1.1.2.5	17
388	PSB	Test Turbine Steam Valves weekly or provide justification for monthly testing.	---	---	---
389	PSB	Provide additional information on short circuit protective devices for circuits that pass through containment penetrations.	---	8.3.1.1.2.11.f)	---
390	PSB	Following maintenance or testing do operating procedures ensure that the diesel generator keep-warm system is properly aligned.	---	9.5.5.4	17
391	PSB	Discuss the lowest barometric pressure anticipated for the geographic area, especially the impact on diesel generator performance.	---	9.5.8.3	---
392	METB	Provide additional information on the release of particulates and iodines from the condenser vacuum pump exhaust.	---	11.3.3	17
393	METB	The turbine building vent stack should contain a radiation monitor and appropriate sampling capability in accordance with Table 1 of SRP 11.5.	---	Refer to Open Item #155	17
394	METB	The normal mode of operation of the recycle evaporators will have the distillate sent from the recycle monitor tanks to the WPS monitor tanks.	---	9.3.4.2.2	17

DSER Open Item No.	NRC Branch	Subject (DSER Section)	Safety Review Question	Affected FSAR Section(s)	Amendment
395	SGEB	Auxiliary/Main Dam Reservoirs	---	---	---
396	SGEB	Rip/Rap Data	---	---	---
397	PSB	Discuss instrumentation/controls on the auxiliary control panel.	---	Remains an Open Item	---
398	PSB	Information needed on the microwave system.	---	Remains an Open Item	---
399	PSB	Clarify electrical separation criteria in the cable spreading area as discussed in FSAR Section 8.3.1, pg. 8.3.1 38.		8.3.1.2.14	11
400	PSB	Information needed on the closure time of the extraction steam reverse current valves.	---	10.2.2	5
401	PSB	Provide a flow path of the Main Steam Isolation Valve and the turbine stop valves.	---	10.3.3	17
402	PSB	Discuss auxiliary lube oil pump controls/alarms.	---	9.5.7.2	17
403	MTEB	Information needed as to whether the new Westinghouse procedures for torqueing and heat treatment of the new guide tube pins are being followed.	---	4.2.2.2.1	17
404	PSB	Clarify the method of color coding class IE cable.	---	8.3.1.2.14.j)	17