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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

Docket No. 50-382

MAY 30 1980

Mr. D. L. Aswell
Vice President, Power Production
Louisiana Power and Light Company
142 Delaronde Street
New Orleans, Louisiana 70174

Dear Mr. Aswell:

SUBJECT: REQUESTS FOR ADDITIONAL INFORMATION - WATERFORD STEAM ELECTRIC STATION, UNIT 3

Enclosed are round one requests for additional information. Included are requests 211.14 through 211.90 concerning reactor systems. Our review of this area was based on the Standard Review Plan and other staff guidance existing prior to the accident at Three Mile Island, Unit 2. The additional requirements associated with that aspect of the review will be the subject of separate correspondence.

Please advise us of the date you expect to provide responses to the enclosed request. If you require any clarification of these matters please contact the staff's assigned project manager.

Sincerely,

A. Schwencer

A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing

Enclosure:
Requests for Additional
Information

cc w/enclosure:
See next page

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ENCLOSURE

REQUEST FOR ADDITIONAL INFORMATION

WATERFORD UNIT 3

DOCKET NO. 50-382

211.0 REACTOR SYSTEMS

211.14 (5.47 9.36) The response to request 211.3 regarding the capability to take the plant to a cold shutdown condition using only safety grade equipment is inadequate. During normal plant operations events which require cooldown of the system and utilization of the RHR system for long-term cooling may occur. For the staff to evaluate the acceptability of the systems and procedures used to achieve a cold shutdown condition, a detailed response in each of the following areas is required:

1. Describe the sequence for achieving a hot shutdown condition within 36 hours, assuming the most limiting single failure. Identify all manual actions inside or outside containment that must be performed and discuss the capability of remaining at hot stand by until manual actions (or repairs) can be performed.

   a. If the steam generator dump valves, operators, air and power supplies are not safety grade, justify how you would cool down the primary system in the event of loss of offsite power and an SSE.

   b. Describe the sequence for depressurizing the primary system using only safety-grade systems, assuming a single failure. Identify all manual actions inside or outside containment that must be performed.

   c. Discuss the boration capability using only safety-grade systems, assuming a single failure. Identify all manual

actions inside or outside containment that must be performed. If the proposed boration method utilizes the charging pumps (assuming a letdown line failure is proposed), provide an evaluation of this approach with regard to concentration of boron source and liquid volume in primary system.

2. Discuss the provisions for collection and containment of RHR pressure relief valve discharge.

3. Describe tests which will demonstrate adequate mixing of the added borated water and cooldown under natural circulation conditions with and without a single failure of a steam generator atmospheric dump valve. Specific procedures for plant cooldown under natural circulation conditions must be available to the operator. Summarize these procedures.

4. Discuss the availability of a Seismic Category I auxiliary feedwater supply for at least 4 hours at hot shutdown plus cooldown to the RHR system cut-in based on longest time for the availability of only onsite or only offsite power and assuming a single failure. If this cannot be achieved, discuss the availability of an adequate alternate Seismic Category I water source.

2.11. (5.4.7 9.3.6) Per the requirements of BTP RSB 5-1, describe the provisions to prevent damage to the RHR system pumps due to overheating, cavitation or loss of adequate pump suction fluid.

2.11.16 (5.4.7 & 9.3.6) Information to verify that the design heat removal capability of the RHR system is consistent with the calculated core decay heat output should be included or referenced in Section 9.3.6 to satisfy the SRP Section 5.4.7 requirements.

2.11.17 (5.4.7 9.3.6) Per the requirement of SRP 5.4.7, manual actions outside the control room must be justified. For all the valves listed in Table 9.3-19 and for the manual valves referenced in the response to question 2.11.3 (5.4.7) provide information to verify that these valves are accessible both during normal operations and under accident conditions. Also provide estimates concerning time required for these actions to be performed.

211.18 (4.6) It is the staff's position that Regulatory Guide 1.68 "Initial Test Program for Water Cooled Nuclear Power Plants" Revision 2 dated August 1978 is applicable to the Waterford 3 plant. Please address the control rod scram timing test program for the Waterford 3 plant relative to the recommendations of Regulatory Guide 1.68 Rev. 2.

211.19 (3.5.1.2) Provide a discussion of credible secondary missiles generated as a result of direct or richochet impact with primary missiles and potential gravity missiles as per the requirement of SRP Section 3.5.1.2 (Rev. 1). With regard to gravity missiles, identify all non-seismic equipment located above the reactor vessel, reactor coolant system piping and components, ECCS piping components, and instrumentation and controls required for ECCS operation or safe shutdown. Provide an evaluation of the consequences of this equipment becoming gravity

missiles and any procedures or controls required to prevent adverse consequences from this occurence.

211.20 (3.5.1.2) Justify not including the following equipment in the discussion of credible missiles:

1. Valves or Valve Stems
2. Pressurizer Heaters
3. Components of Containment Heat Removal System (i.e., fans)
4. Components of Combustible Gas Control System (i.e., pumps)

211.21 (5.2.5) As per the requirements of Regulatory Guide 1.45 and SRP Section 5.2.5 provide the following information:

1. Discuss procedures used by the operator to convert all leakage detection indications in the control room to a common leakage equivalent (e.g., CPM to GPM).

2. Describe the operability testing and system calibration procedures during plant operation for the Waterford leakage detection systems, including the operability checks of the sump monitoring system and the containment fan cooler condensate flow monitors.

3. Provide a schematic diagram for sump level and flow monitoring system.

4. Describe the capability to take grab samples of the containment atmosphere on a periodic basis and manually analyze these samples in a radiochemistry laboratory for particulate activity and to correlate the data to primary system leakage.

5. Discuss the calibration of the air particulate monitor with respect to the isotope being monitored, plate out and decay rate.

6. The containment atmospheric radiation monitor used for the detection of airborne particulate, iodine and gaseous radioactivity is dependent on the containment background levels for detection of a leak. The containment background level is in turn dependent on coolant activity, buildup of activated corrosion products and particulates from the maximum allowable identified leak and normal (expected) unidentified leakage. Show how the sensitivity and response time of the containment atmospheric radiation monitor, employed for monitoring unidentified leakage to the containment, satisfies the requirements of Regulatory Guide 1.45 considering the expected range of containment activity levels stemming from the sources identified.

211.22 (5.2.5) Licensee event reports have identified events where the unidentified leakage has exceeded technical specification limits. In some instances inspection after shutdown has located the source and amount of leakage and it has then been reclassified as identified leakage and the plant returned to power. Evaluate the effect of the super-

position of a spectrum of identified leakage rates on the ability of the unidentified leakage detection systems to meet the sensitivity requirements of Regulatory Guide 1.45.

211.23 (5.2) Your response to request 211.2 is not complete. Please address the following:

1. The electrical, instrumentation and control system must provide alarms to alert the operator to:

   a. properly enable the system at the correct plant condition during cooldown.

   b. indicate if a pressure transient is occurring.

2. To assure operational readiness the system must be tested in the following manner:

   a. A test must be performed to assure operability of the system electronics prior to each shutdown.

   b. A test for valve operability must, as a minimum, be conducted as specified in the ASME Code, Section XI. Justify that, with the application of a single failure to one valve, a single safety relief valve on the shutdown cooling system is satisfactory for overpressure protection. Address an increased testing frequency program for this valve.

3. Provide the results of the analyses performed for the Waterford 3 plant for the most limiting mass and heat input transients.

4. Show that overpressure protection is provided (do not violate Appendix G limits) over the range of conditions applicable to shutdown and heatup operations.

5. Discuss the shutdown and startup plant procedures with respect to the instructions for aligning the SDCS safety relief valves. Specify the temperatures at which the valves will be aligned. Discuss the technical specifications for operability of the valves.

6. Discuss all administrative controls required to implement the protection systems.

7. Provide a description of the relief valve design.

8. Justify that water-solid overpressure transients will not exceed 110 percent of the design pressure of any components in the SDCS.

9. Describe how you have accounted for the flashing of water at the discharge of the safety relief valve in the sizing of the valve or in the overpressure analysis.

10. Discuss the adequacy of the margin between the maximum pressure resulting from an overpressurization event and the setpoint for the automatic isolation of the shutdown cooling system (currently 500 psig).

211.24 (15.1.1.1) (15.1.2.1) Provide a reference to and describe the sensitivity study which shows that the rate of decrease of RCS temperature for the decrease in feedwater temperature transient is less than for the increased steam flow transient.

211.25 (15.1.1.2) (15.1.2.2) Provide a reference to and describe the sensitivity study which shows that the rate of decrease of RCS temperature for the increase in feedwater flow transient is less than for the increased steam flow transient.

211.26 (15.1.1.3) Figure 15.1-6 shows that the pressurizer is emptied during the increased main steam flow transient. Discuss the effect of this occurrence, e.g., will it result in void formation or core uncovery until the level is restored. Provide the bases for your response.

211.27 (15.1.1.3) Figure 15.1-3 for the increased main steam flow transient and Figure 15.1-14 for the inadvertent opening of a steam generator atmospheric dump valve transient infers that the ECCS would be started during these transients. Discuss the potential for filling the system water-solid and resultant operator actions needed to restore a bubble in the pressurizer. Provide an analysis which shows the impact of operator actions and the time frame in which they are

performed. Discuss operating procedure and training for recovering from a rapid loss of the pressurizer bubble.

211.28 (15.1.1.3) (15.1.2.3) Figures 15.1-8, -9, -27 and -28 show a reduction in main steam flow and feedwater flow. Verify that this is accomplished using safety-grade equipment or is a conservative assumption, or provide additional analyses not taking credit for these actions.

211.29 (15.1.2.1) (15.1.2.2) (15.1.2.3) Verify that the single failure assumed for these analyses is the worst-case single failure. It would appear from the results of 15.1.2.3 that this may not be the case, as the assumption of loss of offsite power results in a transient which is less severe than the non-single failure case from a standpoint of RCS temperature and pressure, although this may be offset by the effect of loss of forced reactor coolant flow.

211.30 (15.2) Discuss the applicability of the steam pressure regulator failure transient identified in Standard Review Plan Section 15.2.1 to the Waterford Plant.

211.31 (15.2.1.3) Justify the use of the value of $+0.0 \times 10^{-4}$ for the moderator coefficient given in Table 15.2.2 rather than the value of 0.5×10^{-4} given in paragraph 15.0.3.3.2.

211.32 (15.2.1.4) Provide a reference to the calulations justifying the natural circulation flow rate of 5 percent of nominal full power flow assumed in the analysis.

211.33 (15.0) Provide the initial water volume in the pressurizer used for the various analyses and justify why this value is conservative. Of particular concern are those transients which result in significant pressure increases. In particular, explain why the initial levels shown in Figures 15.2-6 and 15.2-18 are different and if the differences are conservative.

211.34 (15.0) For all analyses of transients with concurrent single failures, provide a reference to the sensitivity study which shows that the failure selected is the worst-case single failure.

211.35 (15.2.2.5) The Standard Review Plan, Section 15.2.7, classified the Loss of Normal Feedwater Flow as an Incident of Moderate Frequency. Provide a justification for the classification here as an Infrequent Incident.

211.36 (15.2.2.5) The loss of normal feedwater flow transient takes credit for operation of the steam bypass system until operator action is taken. It also assumes that the main condenser is available. Address the consequences of this transient, assuming that these non-safety grade systems are not available.

211.37 (15.2.2.5) There is a discrepancy between the value given for the maximum RCS pressure in the test (Section 15.2.2.5.3.3) and Figure 15.2-28 and the value given in Table 15.2-6. Explain or correct this discrepancy.

211.38 (15.2.3.1) Justify the use of the value of 0.0×10^{-4} for the moderator coefficient given in Table 15.2-9 rather than the value of 0.5×10^{-4} given in paragraph 15.0.3.3.2.

211.39 (15.2.3.1) Paragraph 15.2.3.1.3.3 and Figure 15.2-39 give a peak RCS pressure of 2832 psia. Discuss the criteria by which you measured the acceptability of this peak pressure for the feedwater line break event.

211.40 (15.2.3.1) Provide a reference to the sensitivity study which determines the worst case break size, location, plant power level, operating mode, etc. for the feewater line break.

211.41 (15.2.3.2) Justify the use of the value of -2.0×10^{-4} for the moderator coefficient instead of the value of -3.3×10^{-4} given in Section 15.0.3.3.2.

211.42 (15.2.3.2) Address the compliance of the loss of normal feedwater flow with an active failure with acceptance criteria 2.9 of Standard Review Plan 15.2.7.

211.43 (15.2.3.2) Address the concern raised in question 211.26 related to the potential for void formation or core uncovery for the loss of normal feedwater flow with active failure.

211.44 (15.3.2.1) The loss of forced reactor coolant flow analysis assumes operation of the non-safety grade turbine system until the operator takes control. Address the consequences of failure of this system to operate.

211.45 (15.3) Provide an analysis of a reactor coolant pump shaft break as required by Section 15.3.4 of the Standard Review Plan.

211.46 (15.3.3.1) Provide an analysis and discuss the consequences of a single reactor coolant pump shaft seizure assuming the steam bypass system is in the manual mode, operator action is delayed for 30 minutes following the event, and a steam generator safety valve sticks open.

211.47 (15.3.2.2, 15.3.3.2) Provide a justification with basis for the statement that there are no credible single failures which would produce more severe consequences for the loss of forced reactor coolant flow transients.

211.48 (15.4.1.4) Provide a reference to the analysis which shows that a cold shutdown status is the limiting condition for the boron dilution transient.

211.49 (15.4.1.4) An operating PWR has experienced a boron dilution incident due to inadvertent injection of NaOH into the reactor coolant system while the reactor was in a cold shutdown condition. Discuss the potential for a boron dilution incident caused by dilution sources other than the CVCS.

211.50 (15.1.3.1) Reference the sensitivity study which shows that the failure of an HPSI pump to start is the worst case single failure for the steam line break.

211.51 (15.1.3.1) Discuss the effect of emptying the pressurizer during the steam line break. Has the analysis been continued in time long enough to show whether continued shrinkage will result in void formation or core uncovery before the level can be restored with the high head injection system?

211.52 (6.3, 15.0) Certain automatic safety injection signals are blocked to preclude unwanted actuation of these systems during normal shutdown and startup operations. Describe the alarms available to alert the operator to a failure in the primary or secondary system during this phase of operation and the time frame available to mitigate the consequences of such an accident.

211.53 (15.0) Provide a confirmation, with bases, that all transient events would not exceed the acceptance criteria for abnormal operational occurrences when credit is not taken for non-safety grade systems (tubine, trip, turbine bypass, pilot-operated relief valves, etc.)

211.54 (15.0) (6.3) Recent plant experience has identified a potential problem regarding the operability of the pumps used for long-term cooling (normal and post-LOCA) for the time period required to fulfill that function. Provide the pump design lifetime (including operational testing) and compare the continuous pump operational time required during the

short and long-term of a LOCA. Submit information in the form of tests or operating experience to verify that these pumps will satisfy long-term requirements.

211.55 (15.0) Confirm that during the preoperational or startup test phase you intend to verify the valve discharge rates and response times (such as opening and closing times for main feedwater, auxiliary feedwater, turbine and main steam isolation valves, and steam generator and pressurizer relief and safety valves) to show that they have been conservatively modeled in the Chapter 15.0 analyses.

211.56 (6.3, 15.0) Identify the version of the evaluation model used in the LOCA analyses and reference NRC approval letters.

211.57 (15.6.1) Provide an analysis of the inadvertent opening of a pressurizer safety valve as required by Standard Review Plan Section 15.6.1.

211.58 (6.3) The response to request 211.8 indicates that the SIS sump level instrumentation at level -11 ft. MSL is flooded and non-safety grade and that flooding will have no effect on the safety of the plant. There does not appear to be any level instrumentation that would indicate adequate water level (about -4 ft. MSL) in the containment for meeting NPSH requirements prior to manual actual of the recirculation mode.

We require the addition of two (redundant) safety grade containment sump water level indicators with readouts in the control room. The

purpose of these instruments is to provide direct information to the operator thereby reducing the probability of the operator prematurely transferring (manually) from the injection mode to the recirculation mode.

211.59 (6.3.3.3) The staff finds no satsifactory basis for comparing the ECC systems systems of CESSAR, and Calvert Cliffs Unit One, to Waterford 3 in order to assure that small break results for Waterford 3 are bounded by the submitted results of the other plants. Therefore, the staff requires that small break calculations be performed for breaks 0.01 ft^2, 0.05 ft^2, 0.1 ft^2 and 1.0 ft^2 using an approved model.

211.60 (15.6.3.3) Discuss a break of an ECCS injection line. In particular, describe the flow splitting which will occur in the event of a single failure and verify that the amount of flow actually reaching the core is consistent with the assumptions used in the analysis.

211.61 (6.3) If manual operated valve 3SI-V109 A/B failed closed or was inadvertently closed, the FM & EA Item 38 indicates this would result in loss of the miniflow provision for all SI pumps. Justify the lack of redundancy in the miniflow line or modify the design to provide suitable redundancy so that the system safety function can be accomplished, assuming a single failure.

211.62 (6.3) The acceptance criteria in the Standard Review Plan Section 6.3 states the ECCS should retain its capability to cool the core in the event of a single active or passive failure during the long-term

recirculation cooling phase following an accident. Demonstrate that Waterford ECCS design has this capability.

211.63 (6.3) A reported event has raised a question related to the conservatism of NPSH calculations with respect whether the abosolute minimum available NPSH has been considered. In the past the required NPSH has been taken by the staff as a fixed number supplied through the applicant by either the architect engineer or the pump manufacturer. Since a number of methods exist and the method used can affect the suitablility or unsuitability of a particular pump, it is requested that the basis on which the required NPSH as determined be identified (i.e., testing, Hydraulic Institute Standards) for all the ECCS pumps and the estimated NPSH variability between similar pumps including the testing inaccuracies be provided.

211.64 (6.2.2, 6.3) The response to request 211.10 is not adequate. The staff will require experimental verification that the Waterford plant can operate in recirculation without cavitation or air entrainment problems. The experimental program must demonstrate a margin of at least 1 foot of available NPSH over that required for each pump with all pumps at runout or maximum post-LOCA flow.

The test must demonstrate that the design precludes conditions adverse to safety system operation. Test parameters must include: (1) minimum to maximum containment water level, (2) minimum to maximum safety system flow range in various combinations (this includes transients associated with start-up, shutdown, or throttling

of a train or pump), (3) random blockage of up to 50 percent of the screens and grids, (4) approach flow for each dominant direction and combinations thereof, and (5) simulation of break flow or drain flow impinging or originating within line of sight of the sump and its approaches.

If adverse conditions are encountered, the model configuration must be revised until an acceptable configuration is developed and demonstrated to perform over the full range of variables.

The testing may be performed in-plant on a full or reduced scale model of that portion of the plant required to consider the above parameters. A one-third or larger scale model is perferred. If you choose to conduct a scale model test, provide details of the test program. Include information on the model size, scaling principles utilized, comparison of model parameters to expected post-LOCA conditions, and a discussion on how all possible flow conditions and screen blockages will be considered in the model tests. Whenever a reduced scale model is tested, all tendencies for vortex formation must be suppressed. Rotational flow patterns and surface dimples which might be acceptable in full scale tests, probably would not be accepted in a model program. Model testing must include some in-plant testing to demonstrate experimentally that NPSH margin exists for each pump.

211.65 (6.3, 6.2.2) Discuss in detail the extent to which your preoperational test program for the ECCS will conform to the recommendations of Regulatory Guides 1.68,

Revision 2 and 1.79. Specifically, include the procedures which will be used to verify nominal and runout ECCS flow, pump characteristics, piping losses and verification that each check valve in the system is capable of performing both its ioslation and flow function.

211.66 (6.3, 6.2.2) Describe the instrumentation for level indication in the recirculation sump. Also, provide detailed design drawings of the containment recirculation sump including the design provisions which preclude the formation of air entraining vortices during recirculation cooling.

211.67 (5.2.2) Check valves in the discharge side of the high pressure safety injection, low pressure safety injection, RHR, and charging systems perform an isolation function in that they protect low pressure systems from full reactor pressure. The staff will require that these check valves be classified ASME IWV-2000 category AC, with the leak testing for this class of valve being performed to code specifications. It should be noted that a testing program which simply draws a suction on the low pressure side of the outermost check valves will not be acceptable. This only verifies that one of the series check valves is fulfilling an isolation function. The necessary frequency will be that specified in the ASME Code, except in cases where only one or two check valves separate high to low pressure systems. In these cases, leak testing will be performed at each refueling after the valves have been exercised.

Identify all check valves which should be classified categroy AC as per the position discussed above. Verify that you will meet the required leak testing schedule, and that you have the necessary test lines to leak test each valve. Provide the leak detection criteria that will be in the Technical Specifications.

211.68 (5.2.5) Additional information is required for completion of our review of intersystem leakage from the RCS into interfacing systems. Identify all interfacing systems (including systems in addition to those previously identified such as the SDCS, LPIS, and HPIS), and indicate how intersystem leakage will be detected and the sensitivities and response times of the detection systems.

211.69 (5.4) Provide the ASME quality group and seismic category of the main steamline flow venturi.

211.70 (6.3) Discuss the provisions for maintaining the ECCS lines in a filled condition. Describe how water hammer was considered in the design of the ECCS.

211.71 (6.3) Recently, another plant has indicated that a design error existed in the sizing of their RWST. This error was discoverd during a design review of the net positive suction head requirements for the containment spray and residual heat removal pumps. The review showed that there did not appear to be sufficient water in the RWST to complete the transfer of pump suctions from the tank to the

containment sump, before the tank was drained and ECCS pump damage occurred.

It was reported that in addition to the water volume required for injection following a LOCA, an additional volume of water is required to the RWST to account for:

1. Instrument error in RWST level measurements.

2. Working allowance to assure that normal tank level is sufficiently above the minimum allowable level to assure satisfaction of technical specifications.

3. Transfer allowance so that sufficient water volume is available to supply safety pumps during the time needed to complete the transfer process from injection to recirculation.

4. Single failure of the ECCS system which would result in larger volumes of water being needed for the transfer process. In this situation, the worst single failure appears to be failure of a single ECCS train to realign to the containment sump on a low RWST signal. This results in the continuation of large RWST outflows and reduces the time available for the manual recirculation switchover, before the tank is drawn dry and the operating ECCS pumps are damaged.

5. Unusable volume in the tank is present because once the tank suction pipes are reached, the pumps lose suction and any remaining water is unusuable. Additionally, some amount of water above the suction pipes may also be unusuable due to NPSH considerations and vortexing tendencies with the tank.

Preliminary indications are that approximately an additional 100,000 gallons of RWST capacity were needed to account for these considerations. It is our understanding that the design parameters for instrument error, transfer allowance and single failure have changed since the original sizing of the tank.

In light of the above information, discuss the adequacy of your Refueling Water Storage Pool. Provide a discussion of the necessary water volumes to accomodate each of the five considerations indicated above. Justify your choice of volumes necessary to account for each consideration. Provide drawings of your RWSP, showing placement and elevation of tank suction lines, and level sensors. Also, provide operator switchover procedures for aligning to the recirculation mode, with estimates of the time required for each action.

211.72 (6.3) The FM & EA Item 29 indicates plugging of the temporary strainers is a possible common mode failure for the ECCS pumps. Demonstrate that the use of the temporary strainers will not result in a common mode failure of the pumps.

211.73 (6.3) Provide in the technical specifications the range of nitrogren cover gas pressure for the SIT (item 3.5.1.d) and the pump discharge pressure (item 4.5.2.e).

211.74 (6.3) Correct the labels on Figures 6.3-2 and 6.3-3. The indicated valve positions for the shutdown cooling mode of hot leg injection shown in Figures 6.3-6, 6.3-7 and 6.3-9 are either in error or the conditions being represented should be clarified. Table 6.3-4 in conjunction with Figures 6.3-7 and 6.3-8 indicates that the LPSI pumps can be dead-headed or without suction flow. Demonstrate that the ECCS will be operated in a mode which will prevent damage to the pumps.

211.75 (6.3) The FM & EA item 24 indicates there is a potential for loss of both HPSI trains if the RWSP isolation valves are not closed. If the operator fails to close the RWSP isolation valves, demonstrate that the HPIS will continue to adequately cool the core during the recirculation mode.

211.76 (6.3) Provide a time reference for each action in the sequence of action included in the changeover from injection to recirculation. Indicate the time required to complete each action and what other duties the operator would be responsible for at this point in the accident. How much time does the operator have to assure that the system is realigned to the recirculation mode before RWSP water is exhausted if the RWSP isolation valves are not closed? Consider the required pump NPSH in your response.

211.77 (6.3) In Section 6.3.3.8, expand the discussion on specific methods of detecting, alarming and isolating passive ECCS failures during long term cooling to include valve leakage. Show that there is sufficient time for the operator to take corrective action and maintain an acceptable water inventory for recirculation. Justify the basis for the assumed leak rates. Describe how the contaminated water would be handled if one ECCS train must continue to operate with a leak.

211.78 (15.6.3) Provide the basis for the differences in fuel parameters, i.e., peak linear heat rate, gap conductance, fuel centerline temperature and fuel average temperature, shown in Table 15.6-13 and those used in the San Onofre analysis and discuss the effect of these differences on the analysis results.

211.79 (15.3.3.1) The Standard Review Plan, Section 15.3.3 classified the RC pump shaft seizure as an Infrequent Incident. Provide a justification for the classification as a limiting fault.

211.80 (5.2A) In Appendix 5.2A reference is made to a parametric study to determine the design basis accident for sizing the primary safety valves. This study or reference documentation pertaining to this study should be provided. In addition the following information should be provided:

1. SRP 5.2.2 states that the high pressure reactor trip or second safety grade scram signal, whichever occurs later, should be used for sizing the primary side safety valves. The information provided in the FSAR does not satisfy this requirement.

2. Discuss the extent to which the joint industry effort on qualification of reactor coolant system relief and safety valves as developed and managed by the Electric Power Research Institute (EPRI) will be applicable to the Waterford 3 plant. If so, provide certification of and the basis for the applicability of the EPRI program to the Waterford 3 design. This program was presented to the NRC staff in a meeting held on December 17, 1979 and is entitled "Program Plan For the Performance Verification of PWR Safety/Relief Valves and Systems."

   If the program is not applicable provide the basis for safety valve relief requirements considering the discharge of water, steam or a combination of both.

3. Identify and provide the results of the analysis for the most limiting transient used to size the secondary side safety valves. Indicate if in the sizing of these valves the second safety grade scram signal was used. If it was not, a reanalysis should be provided to determine the most limiting transient using this criteria.

4. In the sizing of the valves the interaction of the relieving capacity between the primary and secondary side safety valves should be identified (i.e., for the limiting transient on the primary side the percentage of relieving capacity on the secondary side should be identified).

5. Identify the manufacturer and type (i.e., direct acting, internal pilot operated) or safety valves used on the secondary side and the manufacturer of the primary safety valves.

211.81 (6.3) Identify valves that are required to have power locked out and include under the appropriate Technical Specifications with surveillance requirements listed.

211.82 (6.3) In the event of early manual reset of the safety injection actuation signal (SIAS) followed by a loss of offsite power during the injection phase, operator action may be required to reposition ECCS valves and restart some pumps. The staff requires that operating procedures specify SIAS manual reset not be permitted for a minimum of 10 minutes after a LOCA. Provide the administrative procedures to ensure correct load application to the diesel generators in the event of loss of offsite power following an SIAS reset.

211.83 (6.31) Because of freezing weather conditions, blocking of the vent line on the RWST has occurred on at least one operating plant. Describe design bases and features that preclude this condition from occurring in the Waterford 3 plant.

211.84 (6.3) Provide the following information related to pipe breaks or leaks in high or moderate energy lines outside containment associated with the RHR system when the plant is in a shutdown cooling mode:

1. Determine the maximum discharge rate from pipe breaks for the systems outside containment used to maintain core cooling.

2. Determine the time frame available for recovery based on these discharge rates and their effect on core cooling.

3. Describe the alarms available to alert the operator to the event, the recovery procedures to be utilized by the operator, and the time available for operator action.

A single failure criterion consistent with Standard Review Plan 3.6.1 and Branch Technical Postion APCSB 3-1 should be applied in the evaluation of the recovery procedures utilized.

211.85 (6.3) Consideration should be given to the possibility that local manual valves (handwheel), which could be the source of common mode failure to redundant lines, might be left undetected in the wrong position until a postulated accident occurs. Appropriate administrative controls or valve position indication are examples of methods to be considered to minimize this possibility. Provide a list of all critical manual valves and address the above concern.

Identify all manual valves which have locking provisions.

In addition a recent event (Docket 50-320, LER 78-20/3L, 4/21/78) has brought to our attention that the automatic operation of some motor operated valves can be disabled when the manual handwheel pins are

engaged. Identify all critical motor operated valves associated with Waterford that have this design feature and describe the controls and procedures utilized to prevent the inadvertent disablement of the automatic operation of these valves.

211.86 (15.1.3.1) As explained in Issue No. 1, NUREG-0138, credit is taken for closure of nonsafety-grade valves such as tubine stop valves, control valves, and bypass valves downstream of the MSLIV to limit blowdown of a second steam generator in the event of a steam line break upstream of the MSLIV. In the Waterford Plant design there are various flow paths located between the MSLIV and the turbine stop valves (Figure 10.2-4) that serve various unidentified functions. To confirm satisfactory performance after a steam line break provide the following information, as applicable, related to these various flow paths that branch off between the MSLIV's and the turbine stop valves:

1. the function of the various flow paths and their maximum steam flow
2. the type of valves
3. the size of valves
4. the quality of the valves
5. the design code of the valves
6. the closure time of the valves
7. the actuation mechanism of the valves
8. the closure signal including sensor

9. quality of power sources to valves and sensors

10. quality of air supply to air-operated valves.

In addition, provide justification or analysis that the failure of an MSLIV and the additional blowdown paths result in a less severe accident than that analyzed in Chapter 15.

211.87 (3.6.1.1) Section 3.6.1.1 contains several partial lists of essential systems needed to shut down the reactor and mitigate the consequences of pipe breaks. The containment vessel, steam generators, the steam generator blowdown system, the containment atmospheric release system, and the chemical and volume control system should be listed also. Modify the lists to include all essential systems needed to shut down the reactor and mitigate the consequences of pipe break and show that all these essential systems are protected from postulated piping failures inside the containment.

211.88 (3.6.2) Section 3.6A states "Moderate Energy Piping System analyses are described in Subsections 3.6A.4 and 3.6A.5." Subsection 3.6A.4 is missing and Subsection 3.6A.5 discusses moderate energy piping failures outside the containment. Provide the moderate energy piping analyses for systems inside the containment.

211.89 (3.6.1.2) The nitrogen system should be identified as a high energy system inside the containment and be analyzed for pipe ruptures.

211.90 (5.4.7) Recent plant experience has identifed a potential problem regarding the loss of shutdown cooling during certain reactor coolant system maintenance evolutions. On a number of occasions when the reactor coolant system has been partially drained, improper reactor coolant system level control, a partial loss of reactor coolant inventory, or operating the RHR system at an inadequate NPSH has resulted in air binding of the RHR pumps with a subsequent loss of shutdown cooling. Regarding this potential problem, provide the following additional information:

1. Discuss the design or procedural provisions incorporated to maintain adequate reactor coolant system inventory, level control, and NPSH during partial drain evolutions.

2. Discuss the provisions incorporated to ensure the rapid restoration of the RHR system to service in the event that the RHR pumps become air bound.

3. Discuss the provisions incorporated to provide alternate methods of shutdown cooling in the event of loss of RHR cooling during shutdown maintenance evolutions These provisions should consider maintenance evolutions during which more than one cooling system may be unavailable such as loss of steam generators when the reactor coolant system has been partially drained for steam generator inspection or maintenance.