NUREG-0787
Supplement No. 6

Safety Evaluation Report
related to the operation of Waterford Steam Electric Station, Unit No. 3

Docket No. 50-382

Louisiana Power & Light Company

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

June 1984

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ABSTRACT

Supplement 6 to the Safety Evaluation Report for the application filed by Louisiana Power & Light Company for a license to operate the Waterford Steam Electric Station, Unit 3 (Docket No. 50-382), located in St. Charles Parish, Louisiana, has been prepared by the Office of Nuclear Reactor Regulation of the Nuclear Regulatory Commission. The purpose of this supplement is to update the Safety Evaluation Report by providing the staff's evaluation of information submitted by the applicant since the Safety Evaluation Report and its five previous supplements were issued.

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1 INTRODUCTION AND GENERAL DISCUSSION

1.1 Introduction

On July 9, 1981, the Nuclear Regulatory Commission (NRC) issued a Safety Evaluation Report (SER) (NUREG-0787) related to the operation of Waterford Steam Electric Station, Unit 3. Subsequently, five supplements to the SER have been issued by the staff. This sixth supplement updates the SER by providing the staff's evaluation of information submitted by the applicant (Louisiana Power & Light Company) since the SER and its five supplements were issued.

Each of the following sections of this supplement is numbered the same as the section of the SER that is being updated and the discussions are supplementary to and not in lieu of the discussion in the SER. Appendix A is a continuation of the chronology of the safety review. Appendix B is an updated bibliography. Appendix C contains a copy of the letter from the Advisory Committee on Reactor Safeguards dated October 18, 1983. Appendix D addresses the review of preservice inspection relief requests. Appendix E contains the Federal Emergency Management Agency's interim findings. Appendix F is a list of principal contributors to Safety Evaluation Report Supplement 6 (SSER 6). Appendix I contains a copy of the technical evaluation report on control of heavy loads prepared by Idaho National Engineering Laboratory. The Project Manager is James H. Wilson; he may be reached on (301) 492-7702.

1.7 Summary of Outstanding Issues

Section 1.7 of the SER and its supplements contained a list of outstanding issues. This supplement addresses the resolution of issues previously identified as open. These issues are listed below, along with the section of this report wherein their resolution is discussed.

(1) Review and audit (Safety Review Committee (13.4))
(2) Seismic qualification (3.10)
(3) ICC instrumentation (22, Item II.F.2)

A number of other issues that were reviewed by the staff, primarily as a result of changes to the Final Safety Analysis Report (FSAR), are also closed in this supplement and are listed below, along with the section of this report wherein their resolution is discussed.

(1) Organizational structure and qualifications (13.1)
(2) Training (13.2)
(3) Containment isolation systems (6.2.4)
(4) Quality assurance (17)
(5) Noble gas effluent monitor (22, Item II.F.1)
(6) IE Bulletin 79-27 (7.1.2)
(7) Physical security (13.6)
(8) Thermal-hydraulic design (4.4)

At this time several safety issues remain that have not yet been resolved. These will be addressed in a future supplement to the SER. The following shows these items and the SER section where they are addressed.

(1) Environmental qualification (3.11)
(2) Auxiliary pressurizer spray system (5.4.3)
(3) Operating procedures (22, Item I.C tasks - long term)
(4) Leakage reduction program (22, Item III.D.1.1)
(5) Fire protection (9.5.1)
(6) Liquid and solid radwaste system (11.2)
(7) Initial test program (14)

1.8 Confirmatory Issues

Confirmatory issues are those that were essentially resolved to the staff's satisfaction but for which certain confirmatory information has not yet been provided by the applicant. For the following issues, the staff has received that information and has confirmed the preliminary conclusion.

(1) Piping analysis (3.9.2)
(2) Natural circulation and boron mixing tests (5.4.3)
(3) PSI/ISI (5.2.4, 6.6)
(4) PORV issue (5.4.3)
(5) Seismic qualification (3.10)
(6) Emergency feedwater control (7.3)

At this time two issues remain for which the staff has not yet received the necessary confirmatory information. These issues, which are listed below with the SER sections where they are addressed, will be addressed in a future supplement to the SER.

(1) Control room review (22, Item I.D.1)
(2) Shutdown cooling system relief valves (5.4.3)

1.9 License Conditions

In addition to those issues listed in the SER and its supplements as requiring a license condition to ensure that NRC requirements are met during plant operation, the staff has identified the following license conditions:

(1) During the startup test program, the applicant shall have on each shift a licensed individual with previous startup or operating experience on a comparable PWR, or an advisor who meets these experience requirements.

(2) Prior to exceeding 5% of rated power, the applicant shall complete the following:

   (a) The Parish Plans shall designate, by title, the LP&L official at the Emergency Operating Facility who will have the authority or responsibility to provide protective action recommendations to offsite authorities.

   (b) Letters of agreement with the support parishes, agencies, or political subdivisions of the support parishes, or with other responsible entities, for vehicles and drivers necessary to implement the evacuation plans shall be completed and submitted to the NRC staff.

(c) The Parish Plans shall be amended to specify the vehicles allotted to evacuate the prison population. These vehicles shall have a combined capacity to evacuate the prison population. The plans shall also specify the personnel commitment for drivers and guards. Furthermore, the plans shall clearly indicate that the personnel designated as drivers or guards will have no other emergency duties and the allotted vehicles shall have no other emergency function until after prisoner evacuation is accomplished.

(d) Pickup point information shall be included in the Emergency Broadcast System evacuation message.

In the event that the NRC finds that the lack of progress in completion of the procedures in the Federal Emergency Management Agency's final rule, 44 CFR 350, is an indication that a major substantive problem exists in achieving or maintaining an adequate state of preparedness, the provisions of 10 CFR 50.54(s)(2) will apply.

(3) The applicant shall fully implement and maintain in effect all the provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to the authority of 10 CFR 50.54(p). The approved plans, which contain safeguards information as described in 10 CFR 73.21, are entitled "Site Security Plan Waterford Steam Electric Station Unit No. 3," Revision 6 dated July 6, 1981, Revision 7 dated February 21, 1983, Revision 8 dated April 10, 1984, transmittal letter dated April 11, 1984; "Waterford 3 Steam Electric Station Safeguards Contingency Plan," dated February 1, 1980 as revised July 1, 1980, Revision 2 dated March 14, 1983, and Revision 3 dated January 16, 1984, transmittal letter dated January 12, 1984; "Waterford Generating Station Guard Training & Qualification Plan," dated February 1, 1980, as revised by pages submitted by letter dated April 23, 1981, Revision 2 dated December 19, 1983, transmittal letter dated December 12, 1983.

(4) Prior to startup following the first refueling outage, the applicant shall have made commitments acceptable to the NRC regarding the guidelines of Sections 5.1.2 through 5.1.6 of NUREG-0612 (Phase II).

(5) In order to protect the control room against toxic gas hazards at the Waterford 3 site, the applicant shall complete the following:

(a) Within 6 months after issuance of a full-power license, the applicant shall demonstrate a detection system capable of detecting and indicating the presence of toxic gases at the control room air intakes. The toxic gases to be detected are sulfur monochloride, thionyl chloride, acrylonitrile, acrolein, benzene, ethylene oxide, sulfur dioxide, and hydrogen chloride. Should the applicant fail to demonstrate an operable system within this time with respect to any of the above gases, an alternate protective measure (e.g., special procedures and continuous surveillance) shall be proposed and submitted for staff review.

(b) Prior to startup following the first refueling outage, the applicant shall have an installed, operable broad-range detector system for detection of toxic gas hazards and shall propose associated Technical Specifications for that system for inclusion in Appendix A of the license.

1.13 Independent Design Verification Program

Background

Torrey Pines Technology (TPT) was engaged by the applicant to conduct an Independent Design Verification Program (IDVP) review of the emergency feedwater (EFW) system for Waterford 3. The proposed scope of the IDVP was presented by TPT to the NRC staff at a meeting held on August 26, 1982. Following staff comments at the meeting, TPT subsequently revised the review scope by letter dated September 1, 1982, to include the changes recommended by the staff for review of the EFW system. NRC staff approved the revised scope by letter dated September 28, 1982, which included a technical review of the EFW system design to determine if the design control process converted the design basis of the EFW system into an adequate design. In addition, the program included a physical verification of the conformance of the as-built system to the requirements of the design documents. The objective of the program was to provide increased assurance that the station was properly designed and constructed.

The program included the following six tasks:

Task A - design procedure review - evaluation of compliance of design procedures and control with NRC-approved quality assurance (QA) program

Task B - design procedure implementation review - evaluation of compliance of EFW system design documents with the established design procedures and controls identified in Task A

Task C - technical review - evaluation of EFW system design to determine if system is adequate to perform its intended function

Task D - physical verification of selected portions of EFW system to establish conformance of the installation to the requirments of the design documents and specifications

Task E - processing of potential findings, including evaluation of applicant's response and followup

Task F - program management, administration, and reporting

TPT Evaluation Process

TPT assigned specific items to technical personnel that were to be reviewed in accordance with documented procedures. When a reviewer uncovered an apparently significant deviation, a Potential Finding Report (PFR) was prepared. After the PFR was processed completely, the applicant proposed corrective action for each valid PFR. TPT reviewed the applicant's proposed action for each PFR that could conceivably result in a substantial safety hazard, and, if required, iterations were made until TPT found the proposed corrective action acceptable.

Assessment by TPT

During the evaluation, TPT reviewers generated 38 PFRs. Table 1.1 lists the PFRs, their subject, which task they were identified under, and their final classification. Subsequent processing of these PFRs showed that 14 were invalid (i.e., additional information showed no deviation exists), 20 were judged by TPT as having no potential to cause a substantial safety hazard, and 4 were judged by TPT as having the potential to cause a substantial safety hazard. The apparent deviation in each of these four PFRs was called a finding. A corrective action was developed for each finding and accepted by TPT. These four findings are summarized in Table 1.2.

Assessment by NRC Staff

The NRC staff reviewed the TPT evaluation reported in Technical Report GA-C16900 dated March 1983 and met with TPT and Louisiana Power & Light Company representatives to discuss the design verification results on June 9, 1983. The TPT report includes a technical summary, the program results, and a compilation of the 38 PFRs, corrective action plans for each of the 4 findings, and the TPT review of these corrective action plans.

1.13.1 Design

1.13.1.1 Systems

The EFW system at Waterford 3 is designed to supply an independent source of water to the steam generators during accident and transient conditions in the event of a loss of the main feedwater supply. The major components of the EFW system are three essential safety-grade pumps: one 700-gpm (nominal) steam-driven pump and two 400-gpm (nominal) motor-driven pumps. The EFW system water supply is provided by the condensate storage pool with a backup supply available from the wet cooling tower basins. The system provides two redundant flow paths (one to each steam generator), each flow path containing redundant active components.

During the review of the EFW system performance calculations, TPT initially established a concern in PFR-001 that the EFW pumps are not capable of providing 700-gpm design flow against a steam generator pressure (1,227.5 psig) that corresponds to the maximum steam safety valve set point plus tolerances and accumulation as required in the design criteria. Later review by TPT of FSAR Chapter 15 and the Combustion Engineering (CE) Balance-of-Plant Manual indicated that the EFW system is required to provide a total of 700-gpm flow against a steam generator pressure of 1,100 psia under all postulated accident conditions. Further, the pump performance curve shows that the above design EFW flow is delivered at the pressure used in the FSAR accident analyses. On the basis of the above, TPT classified PFR-001 as invalid. The staff agrees with TPT's resolution of this concern.

EFW pump suction flow from the condensate storage pool is normally provided through two separate lines. These lines connect to a common suction header, which supplies all three pumps. However, during the review of the EFW system performance calculation, TPT observed that there was no consideration given to EFW system operation with only one suction line available from the condensate storage pool. Therefore, TPT performed an independent calculation to determine if adequate net positive suction head (NPSH) is available at the maximum pump discharge flow with only one suction line available. PFR-003 was developed when this calculation indicated that if there is suction flow to the three EFW pumps through only one of the two supply lines from the condensate storage pool and if the pump discharge flow is at the maximum identified value (1,915 gpm at runout conditions for three pumps) rather than the design value, the potential for cavitation of a pump could exist because of inadequate margin in NPSH available. However, the EFW system is required to provide 700 gpm to the steam generators at 1,100 psia pressure. On the basis of their review, TPT classified this PFR as an observation not requiring further applicant action because (1) while the maximum flow to the steam generator is 1,915 gpm, the design flow is 700 gpm, and there is ample NPSH available to deliver the design flow with one suction line (46 ft available compared with 17 ft required); (2) there is approximately 10% margin in NPSH available to deliver the maximum flow; and (3) even if a pump is damaged by cavitation, the remaining two EFW pumps have adequate capacity and NPSH available with only one suction line to deliver the design flow. The staff concurs with the above resolution.

During the review of FSAR Sections 7.3.1.1.6 and 10.4.9.2 and Table 15.2-8, TPT observed that the EFW pumps are started by a signal from the engineered safety features actuation system and reach full speed in about 4 sec. However, the isolation valves remain closed for another 50 sec before they open in response to a steam generator low-low level signal. Review of the EFW arrangement drawings and a field walkdown indicated that from the pumps the EFW discharge lines rise to an elevation which is 49 ft above the top of the condensate storage pool before running horizontally for approximately 90 ft and then downward toward the EFW isolation and control valves. Check valves are also located in the pump discharge lines upstream of the flow control and isolation valves. TPT's discussion with Anchor-Darling, the check valve supplier, indicated that check valves would be expected to leak more than the isolation valves. If the water drained out of the discharge lines through the leaking check valves toward the pumps faster than water leaks into the lines through the isolation valves, a vapor pocket could occur in the lines, and waterhammer damage could result on pump startup. Further, the piping drawing shows no provision to ensure that the EFW lines are maintained full of water. TPT also performed an independent calculation of the stresses on the EFW piping and supports subjected to waterhammer. The piping analysis showed that the EFW system piping will be overstressed and could be damaged because of waterhammer. On the basis of the above, TPT developed PFR-006, which identified a concern regarding the potential for waterhammer in the EFW pump discharge lines.

The staff concurs with TPT that EFW system piping might be damaged because of waterhammer if the piping downstream of the pump is not full of water. This has occurred in other operating nuclear power plants and is documented in NUREG-0582, "Water Hammer in Nuclear Power Plants." Because TPT believed this concern has implications in other fluid systems, it was classified as a finding and subsequently the entire Waterford 3 piping design was reviewed by the applicant for similar waterhammer potential. This review did not identify any other safety-related piping configuration where stagnant water could leak through valves so that a condition resulting in waterhammer might occur. This waterhammer investigation was performed on the circulating water system, feedwater system, and other systems with fast closure valves, such as main steam, safety injection, and pressurizer safety/relief valves.

To accommodate the potential for check valve leakage over the plant life, the applicant revised the design of the EFW discharge piping to include a 1-in. bypass line with a 1/4-in. orifice around the EFW system control and isolation valves in each loop to ensure that the pump discharge lines remain full of water and thus prevent voiding that may lead to waterhammer. The applicant committed to implement this design change before fuel loading. To prevent a recurrence of this concern, the applicant committed to review all future piping configuration changes before installation for possible leakage through valves to preclude the potential for waterhammer. On the basis of the above, the staff concludes that the implementation of this corrective action plan satisfactorily resolves this finding.

In PFR-017, on the basis of a visual examination of the field installation, TPT determined that there is a possibility of water collecting in the steam supply lines to the turbine-driven feedpump with resulting danger of waterhammer on turbine startup. TPT was concerned that leakage of main steam through two 6-in. isolation valves during normal operation would be greater than leakage through a single available 2-in. drain valve. TPT classified this PFR invalid when it was determined during discussion with the applicant that any condensation that forms in the piping will drain out through the steam trap, which is continually available for draining. If condensation exceeds the capacity of the trap, a normally closed motor-operated valve would open to provide additional drainage capacity. During normal plant operation, the steam supply lines to the turbine-driven pump downstream of steam isolation valves are at atmospheric pressure. Because the steam lines are maintained at 450°F by heat tracing, any leakage across the isolation valves would not condense, thus precluding the formation of water slugs. The staff concurs with TPT that this concern has been satisfactorily resolved.

In PFR-028, TPT noted that turbine performance data including test certificates for the EFW pump turbine were not available at the time of the verification walkdown. This concern was classified as an observation because the lack of performance curves did not create a safety hazard since the turbine performance was established during startup testing. However, to complete the documentation, the applicant issued Design Change Notice ME-34, dated February 2, 1983, to document the turbine data. The staff concurs with the above resolution.

TPT undertook a verification of the applicant's design approach to confirm that design information is properly implemented. Planning logic networks (PNLs) are normally used to identify interface reviews of design information in accomplishing the above. The PLN provides a systematic method for scheduling transmittal of major design information as work progresses. In PFR-029, TPT stated that the PLN, or an acceptable alternative method such as the planning and control system, which depicts in a logical way how the overall design should proceed, was not available during the audit at Combustion Engineering (CE). This PFR was classified invalid when the PLN was retrieved from stored records and was made available for the TPT audit. TPT states that the PLN meets the requirements as stated in the CE QA manual and was found satisfactory. The staff concurs with the above resolution.

On the basis of its review of the IDVP report and the implementation of the corrective actions described above, the staff considers that the independent design review of the Waterford 3 EFW system has acceptably verified the functional adequacy of the system. The staff further concludes that the results of the IDVP do not alter its previous safety evaluation for Waterford 3.

1.13.1.2 Structural

The staff has reviewed those PFRs concerning structures that have been classified as findings or observations to assess whether the staff could reasonably reach conclusions similar to those made by TPT.

PRF-013 identified incorrect bending moment calculations for the 3-ft-thick reinforced concrete wall associated with the condensate storage pool. These errors resulted in underestimating the required area of reinforcing steel within the wall.

The applicant has revised the incorrect calculations to eliminate the three errors identified in PFR-013. However, at the same time the applicant has modified previous overconservativeness in the calculations for the determination of settlement. The combined effect of correcting the three calculational errors and modifying the values of relative displacement of floors, which reduced the applicable bending moment, shows that adequate reinforcing steel has been provided for the 3-ft-thick wall (#9 @ 12 in.).

PRF-014 identified incorrect calculations for the condensate storage pool support beam because of the omission of the loading effects of a 2 ft 0 in. wall that transmits loads to the subject beam. The applicant has revised his calculations to include the weight of this wall, to reflect the as-built condition, but at the same time eliminated an overconservative assumption in the computation of the maximum moment and shear. The combined effects resulted in an adequate beam design with adequate reinforcing steel. In addition, more conservatism was identified as a result of the practice of considering the weight of fluid that is replaced with concrete and accounting for it in the summation of the loads.

PRF-016 identified that EBASCO was unable to locate calculations for the air handler supports. EBASCO performed a reevaluation of the existing design for the air handler supports for review by TPT. The calculations demonstrated the adequacy of the existing design.

TPT classified PFR-013 and -016 as observations and PRF-014 as a finding. The staff agrees with the conclusion of TPT regarding the classification of the above three items and the adequacy of the existing designs for the reinforced concrete wall, the reinforced concrete beam, and the supports.

1.13.1.3 Mechanical

The staff has reviewed those PFRs concerning piping and components that have been classified as findings or observations to assess whether the staff could reasonably reach conclusions similar to those made by TPT that the PFRs were classified properly.

In PFR-005, TPT found that the feedwater isolation and control valves were located outdoors, whereas the design specifications stipulated the valve location to be indoors. TPT was concerned that the valves might not be qualified for the outdoor colder temperatures and environmental consequences. TPT classified the PFR as an observation after (1) the valve vendor (Masoneilan) stated that the as-installed position of the valves (upright vertical operator), including all accessories, can be outdoors without affecting their safety function and

(2) the design specification was changed to identify the location as being outdoors with the valve vendor providing a Certificate of Conformance that the valves and accessories will function in the as-installed outdoor environment. The staff believes that this PFR was appropriately classified as an observation.

In PFR-007, the piping design specification for load combinations was found to be inconsistent with FSAR commitments. However, the design specification was found to be in error and was subsequently revised to be consistent with both FSAR commitments and acutal design practice. The staff concurs with the TPT classification of this PFR as an observation and does not believe that this concern extends into other systems.

In PFR-008, TPT found that as-built loads greatly exceeded the design loads for a piping support, and consequently, the stresses in the support attachment plate exceeded the allowable values. However, during the construction process, the support was relocated closer to the centerline of the main structural beam. The effect of the increased loading and the reduced eccentricity offset each other, and the plate stresses were shown to be within the allowable values. It should be noted that the revised calculation that accounted for the construction change was performed after the PFR was issued. However, the architect-engineer (EBASCO) was "caught in the middle of changing (the) analysis." Apparently, the as-built reconciliation of support loads was not completed at the time of the TPT review. The staff concurs with the TPT classification of this PRF as an observation but has difficulty in concluding the as-built reconciliation process for support loads was effective because it was not completed at the time of the TPT review. This will be discussed further in the staff evaluation of Task D.

In PFR-011, TPT identified a potential finding concerning the classification of local bending stresses at snubber lugs on the steam generator. The local bending stresses resulting from seismic loadings were combined with primary stresses resulting from normal operation plus the operating basis earthquake and were evaluated to 3.0 Sm (primary plus secondary stress limits). TPT contends that the local bending stresses resulting from seismic loadings should be classified as primary stresses, and thus, the stress limit should be 1.5 Sm (not 3.0 Sm). TPT classified this PFR as an observation when it was shown that if normal operating pressure rather than design pressure is used, the combined stresses will be within the 1.5-Sm limit.

The staff discussed this item with TPT because it was concerned about the generic implications of this item. If CE evaluates local bending stresses resulting from seismic loadings as secondary stresses, as identified in PFR-011, then there could be other CE Class 1 components for which the same assumption was used. Consequently, the calculated primary stresses have the potential of exceeding the allowable stress value by a factor of two.

After the discussions, TPT again reviewed the data relating to the classification of stresses by CE for other components reviewed by TPT in their Independent Design Verification Program and documented this reevaluation in a letter from F. D. Carpenter to G. Knighton dated August 17, 1983. The portion of CE-designed components selected for the rereview included the steam generator upper support lugs, feedwater nozzle, lower support skirt, and the sliding base. TPT found that the method used to evaluate stresses in the upper support lugs (identified in PFR-011) was not used for the other CE components. TPT determined that the stresses for these other components were classified properly and the proper stress allowable value was applied. The potential finding identified in PFR-011 was verified by TPT to be an isolated case based on a specific method, and no other instances were found.

On the basis of the TPT reevaluation, the staff concurs that PFR-011 was correctly classified as an observation and the concern does not extend into other areas.

In PFR-015, the turbine nozzle loads were found to exceed the vendor's allowable values. The PFR was classified as an observation when (1) the turbine vendor documented that increased nozzle allowable values were permitted and (2) the attached piping was rerouted. The staff agrees with TPT that, with respect to ensuring that vendor allowable nozzle loads have been met, PFR-015 can be classified as an observation. However, with respect to ensuring that condition identification work authorizations (CIWAs) have been properly dispositioned, the concern could extend to other CIWAs. TPT concluded that the proper dispositioning of CIWAs is uncertain because this was the only one reviewed.

Under Task D, "physical verification procedure," several PFRs were issued which identified discrepancies between design drawings and as-installed items. TPT did not identify any findings under Task D and stated that "it was judged that the dimensional discrepancies would have been found and resolved by EBASCO prior to completion of construction." This was based on the fact that EBASCO has a procedure for correlating a final walkdown with a final stress analysis review. Furthermore, TPT concluded that "it is judged that the installation of the selected portions of the Waterford-3 EFW System will conform to the requirements of the design documents."

The staff believes that because the piping and pipe support installation had not been fully completed at the time of the TPT walkdown, it cannot be concluded with definitive assurance that the as-built piping and supports will conform to the design requirements. In hindsight, it might have been more appropriate to have selected a system for which the as-built reconciliation process had been completed. However, the staff concurs with the TPT report that the discrepancies that were found would not have resulted in a significant safety concern and that the existing procedures for reconciling the as-built condition with the design documents would likely have identified the dimensional discrepancies.

1.13.2 Quality Assurance

TPT concluded that the QA program including QA procedures and specific design control procedures for the EFW system were adequate and responsive to 10 CFR 50, Appendix B, and the commitments in the Preliminary Safety Analysis Report.

The staff reviewed the 38 PFRs and concludes that TPT's assessment of each PFR was acceptably conservative. That is, the staff agrees that the apparent deviations in 34 of the PFRs were either invalid or have no potential to cause a substantial safety hazard.

The independent design review of Waterford 3 by TPT indicated that the QA program, design process, and procedures for the EFW system are acceptable except for four findings for which appropriate corrective actions have been described. The IDVP, although by no means a comprehensive review of the entire Waterford 3 plant, was used to provide a measure of assurance that the facility has been designed and constructed properly. Further review of the QA/QC activities at Waterford 3 have been undertaken by the staff and results of that review will be presented in a subsequent supplement to the SER.

1.13.3 Corrective Actions

The corrective action plans for the four findings, which were proposed by Louisiana Power & Light Company and approved by TPT, are acceptable to the staff and their implementation will be verified by inspectors from Region IV.

Table 1.1 List of Potential Finding Reports (PFRs)

PFR* no.	Task	Subject	Classification
001	C	Feedwater pumps not capable of 700-gpm flow against steam generator pressure	Invalid
002	C	Steam pressure could exceed ASME Code allowables	Observation
003	C	Inadequate pump net positive suction head with flow through one suction line and maximum identified pump discharged	Observation
004	C	Inadequate specification of humidity requirement	Observation
005	C	Valve location questioned - specified to be indoors	Observation
006	C	Potential for waterhammer exists because of water leakage through valves	Finding
007	C	Piping design specification not consistent with FSAR	Observation
008	C	Piping support stress resulting from as-built piping loads exceeds FSAR limit	Observation
009	C	Potential for pipe freezing exists in outdoor piping	Finding
010	C	As-built piping loads on steam generator nozzle greater than load analyzed by Combustion Engineering (CE)	Invalid
011	C	Unconservative classification of stress category	Observation
012	C	No seismic load in X direction for steam generator support skirt and sliding base	Invalid
013	C	Incorrect bending moment calculations for condensate storage pool wall	Observation
014	C	Incorrect calculations for condensate storage pool support beam	Finding

*The actual numbering format of PFRs was PFR2448-xxx, which is shortened here to the last three digits.

Table 1.1 (Continued)

PFR* No.	Task	Subject	Classification
015	C	Turbine nozzle load exceeds manufacturer's requirements	Observation
016	B	Original calculation missing for air handling unit supports	Observation
017	C	Potential for waterhammer in steam supply line to turbine for pump A/B	Invalid
018	C	Equipment specification does not include requirements for radiation dose qualification	Finding
019	D	Vent location not per drawing	Observation
020	D	Drain line location not per drawing	Invalid
021	D	Piping not installed per drawing	Observation
022	D	Piping not installed per drawing	Observation
023	D	Piping incorrectly identified and not installed per drawing	Observation
024	D	Interfacing services to valve not connected	Invalid
025	D	Support location not per drawings	Observation
026	D	Support location not per drawings	Invalid
027	D	Support location not per drawings	Observation
028	D	Steam turbine data not available	Observation
029	B	Project logic networks (PLNs) did not describe EFW design inputs or interfaces	Invalid
030	B	CE's balance-of-plant document not reviewed/approved and marked tentative	Observation
031	B	Bergen-Paterson (B-P) Pipe Hangers design/analysis document incomplete - no checks or approvals	Invalid
032	B	Current issue of specification not available in B-P's home office	Invalid

*The actual numbering format of PFRs was PFR2448-xxx, which is shortened here to the last three digits.

Table 1.1 (Continued)

PFR* No.	Task	Subject	Classification
033	B	B-P revision/issue control lacking and requirement missing from project instructions	Invalid
034	B	Procurement document control requirements not suitable	Invalid
035	B	Design document revision/issue not current	Invalid
036	B	Drawing update process of approximately five field change requests or 1 year violated	Observation
037	D	Installed piping and pipe supports not consistent with as-built stress analysis	Observation
038	C	Tornado load not analyzed for EFW piping and supports located outdoors	Invalid

*The actual numbering format of PFRs was PFR2448-xxx, which is shortened here to the last three digits.

Table 1.2 Corrective action plans

PFR no.	LP&L Corrective Action Report
006	Waterhammer on EFW system piping
009	Valves and piping subject to freezing
014	Concrete beam design deficient (repetitive errors in calculational logic)
018	Equipment radiation dose qualification (repetitive failures to implement FSAR requirements)

2 SITE CHARACTERISTICS

2.2 Nearby Industrial, Transportation, and Military Facilities

2.2.1 Nearby Industries

In Supplement 2 of the SER, the staff indicated that the applicant at its request had made a number of commitments regarding toxic gas protection of the control room operators. Specifically, the applicant had committed to chlorine and ammonia detectors, broad-range toxic gas detectors, a hotline communication system between Waterford 3 and the St. Charles Parish Emergency Operations Center (EOC), a periodic survey of the local industrial and transportation activities, and letters of agreement with local industries for notification of toxic chemical inventory changes. The above commitments have been carried out by the applicant.

In terms of hardware, the applicant has installed chlorine and ammonia detectors and is in the process of installing broad-range detectors. The applicant has not been able to demonstrate an operable broad-range toxic gas detector (BRTGD) system. The applicant has established a hotline communication with the St. Charles Parish EOC and has implemented a control room operator and plant personnel training program and procedures with respect to awareness, human detection, and response for toxic gases. The staff considers these measures to be adequate for current plant operation (less than 6 months from issuance of the full-power license). In the context of a projected plant lifetime, however, the staff believes that an additional degree of protection in terms of the BRTGD system is appropriate. For example, it is conceivable that at some time within the projected plant lifetime, the EOC could become unavailable for reasons beyond the applicant's control. If this did occur, the presence of the BRGTD system would provide the necessary control room protection until the availability of the EOC or some alternate toxic gas warning arrangement could be provided. Similarly, considering the relatively high density of hazardous materials in the area, the staff believes that within the lifetime of the plant, other unforeseen developments could impact one or more of the control room protection measures. Hence, the staff believes that for long-term operation, the provision of the BRTGD system is necessary. On this basis, it is the staff's position that an operable BRTGD system, or its equivalent, will be necessary beyond the initial 6-month plant operation time to make the continued operation of the plant acceptable to the staff. Specifically, beyond 6 months after the issuance of the full-power license, the applicant shall demonstrate a detection system capable of detecting and indicating the presence of toxic gases at the control room air intakes. The toxic gases to be detected are sulfur monochloride, thionyl chloride, acrylonitrile, acrolein, benzene, ethylene oxide, sulfur dioxide, and hydrogen chloride. Should the applicant fail to demonstrate an operable system within this time with respect to any of the above gases, additional protective measures (e.g., special procedures and continuous surveillance) shall be proposed and submitted for staff review. In any event, before startup following the first refueling outage, the applicant shall have installed an operable broad-range detector system and shall propose associated Technical Specifications for that system for inclusion in Appendix A of the license. The proposed Technical Specifications will be reviewed by the staff to ensure that GDC 19 and SRP Section 6.4 (NUREG-0800) guidelines are met. The above measures will constitute a license condition for the applicant.

In view of the above consideration and on the basis of the requirements to be met by the applicant, the staff considers that the Waterford 3 control room habitability system meets GDC 19 with respect to toxic gas protection.

With respect to periodic surveys of the local industry, the applicant has committed to conducting a survey every 4 years. In addition, the applicant has obtained letters of agreement from the area industries regarding the notification of significant product line changes. The staff finds that the above provisions adequately address the toxic gas hazards with respect to Waterford 3 and considers this issue resolved.

The staff has been informed by a letter dated November 30, 1983, that the applicant is proposing to construct a 24-in. pipeline for carrying process steam from Waterford Units 1 and 2 to the Union Carbide plant. A portion of the proposed pipeline would pass within the Waterford 3 exclusion zone and at its closest approach would be about 2,700 ft from the reactor center.

The applicant states that the line is expected to operate at a pressure of 650 psig or less and a temperature of 800°F. The pipeline will be supported 4 ft above ground on supports spaced approximately every 40 ft.

The staff has reviewed the applicant's analyses with respect to the steam jet impingement, pipe whipping, and thermal hazards. These effects were considered with respect to the overhead transmission lines, the rail traffic on the Texas and Pacific railroad, and the nearby 10-in. natural gas pipeline. There are no other safety-related facilities or activities in the immediate vicinity of the proposed pipeline. The staff concurs with the applicant's findings that the location of the proposed pipeline, in conjunction with the pipeline supports, precludes any significant hazard to either the overhead transmission lines or the rail traffic. Although it is possible that a steamline rupture could damage the natural gas pipeline, the potential release of natural gas would not pose a significant hazard to Waterford 3. The potential effects of a break in the natural gas pipeline are bounded by those analyzed for larger diameter pipelines at closer distances to the plant. The analyses indicated that the hazards from the natural gas releases for these pipelines presented no hazard to the plant.

The staff finds that the proposed process steamline does not pose a significant hazard to the Waterford 3 plant.

2.2.4 Airports

In Supplement 5 of the SER, the staff noted that the Federal Aviation Administration (FAA) was in the process of considering for approval a general aviation airport within 5 mi of the Waterford 3 plant site.

The staff has discussed the status of the airport plan with the applicant. The applicant indicated that the proposed airport sites met sufficient opposition from local interests (e.g., industry and residents) so that the plan had been deferred. However, more recently, efforts have been made to reinitiate the plan. It is the staff's understanding that the resolution of the plan involves significant Federal action, including the issuance of an environmental impact statement (EIS). The EIS comment period will provide an opportunity for the staff, as well as the applicant, to furnish input regarding any proposed airport site as it affects the safety of Waterford 3. During past reviews the staff's concerns regarding aircraft hazards to nuclear power plants have been weighted heavily by the FAA and were instrumental in affecting the choice of an airport site. On this basis, the staff believes that the FAA will be receptive to its safety concerns and will not approve any site that would create a significant hazard with respect to a nuclear plant.

The staff plans to follow the development of this issue and conduct a safety evaluation with respect to any significant action that may occur on this matter.

3 DESIGN CRITERIA - STRUCTURE, COMPONENTS, EQUIPMENT, AND SYSTEMS

3.9 Mechanical Systems and Components

3.9.2 Dynamic Testing and Analysis of Systems, Components, and Equipment

3.9.2.1 Piping Preoperational and Startup Testing Program

In Section 3.9.2.1 of the SER, the staff identified a confirmatory item regarding the piping preoperational and startup testing program. The purpose of the tests is to ensure that the piping vibrations are within acceptable limits and to verify that the piping systems can expand thermally in a manner consistent with the design intent. In a letter dated June 8, 1983, the applicant provided a summary of the thermal testing of piping systems based on the preoperational phase of the Waterford 3 startup testing program. The staff has reviewed the results of the testing program and finds that the test procedures and the test results submitted by the applicant provide a reasonable basis for satisfying the confirmatory item in Section 3.9.2.1 of the SER. Thus, the staff considers the confirmatory item regarding the piping preoperational and startup test program closed.

3.10 Seismic and Dynamic Qualification of Seismic Category I Mechanical and Electric Equipment

The staff's evaluation of the adequacy of the applicant's program for qualification of safety-related electrical and mechanical equipment for seismic and dynamic loads consists of (1) a determination of the acceptability of the procedures used and the standards followed and the completeness of the program in general and (2) an onsite audit of selected equipment items to develop the basis for the staff judgment on the completeness and adequacy of the implementation of the entire seismic and dynamic qualification program.

The Seismic Qualification Review Team (SQRT) reviewed the equipment dynamic qualification information contained in the pertinent Final Safety Analysis Report (FSAR) Sections 3.9.2 and 3.10 and made a site visit on September 15 through 18, 1981, to determine the extent to which the qualification of equipment as installed in Waterford 3 meets the current licensing criteria as described in Institute of Electrical and Electronics Engineers (IEEE) Std 344-1975, Regulatory Guides (RGs) 1.92 and 1.100, and the Standard Review Plan (SRP) Section 3.10. Conformance with these criteria are required to satisfy the applicable portions of GDC 1, 2, 4, 14, 18, and 30 (Appendix A to 10 CFR 50); Appendix B to 10 CFR 50; and Appendix A to 10 CFR 100. Representative samples of safety-related electric and mechanical equipment, as well as instrumentation, included in both nuclear-steam-supply-system (NSSS) and balance-of-plant (BOP) scopes, were selected for the plant site review. The review consisted of field observations of the actual equipment configuration and its installation followed by the review of the corresponding test and/or analysis documents. Because of the large number of unresolved issues resulting from the first SQRT audit, a second SQRT audit was conducted. The second audit sample of safety-related mechanical and electric equipment was different from the sample selected for the first SQRT audit, which was conducted on August 31 through September 3, 1982. On the basis of the second SQRT audit, the SQRT concluded that the applicant has expended a great deal of effort in improving his seismic and dynamic qualification program since the first SQRT audit, as evidenced by the significant improvement observed during this audit. Most of the equipment-specific concerns identified from the first and second audits were resolved during the second SQRT audit. However, some generic and equipment-specific concerns remained following the second site visit. In response to these concerns, the applicant provided submittals following the second audit. These submittals were provided on January 21 and 27, 1983, February 1, 2, 8, and 11, 1983 and February 23 and March 2, 1984. The staff's concerns as identified in Supplement 1 and their corresponding disposition on the basis of these submittals by the applicant are summarized below.

Status of Generic Concerns

(1) The applicant was to (a) identify equipment not covered by his previous justification for single frequency and/or single direction test methods and (b) provide additional justification for the qualification of equipment for which the containment floor horizontal response spectra are applicable. The applicant informed the staff in his February 11, 1983, submittal that an evaluation of the seismic qualification of all such equipment had been made and that none of this equipment was qualified by the single frequency method. The staff considers this response acceptable and this concern is closed.

(2) The applicant was to address the effect of aging on the seismic capacity of equipment located in the mild environment. The applicant has provided a description of a surveillance and maintenance program that has been established in lieu of actually aging equipment before seismic testing. This program description is contained in the applicant's February 8, 1983, and February 23, 1984, submittals. Examples from the surveillance and maintenance program have been provided. The staff considers this response acceptable and this concern is closed.

(3) The applicant was to provide additional justification for the validity of seismic qualification of complex electric equipment by analysis only.

For qualification for equipment operability, the acceptance criterion of SRP Section 3.10 states that tests and analyses are required to confirm the operability of all mechanical and electrical equipment during and after an earthquake of magnitude up to and including the operating basis earthquake (OBE) and safe shutdown earthquake (SSE), and for all static and dynamic loads from normal, transient, and accident conditions. Before SSE qualification, it should be demonstrated that the equipment can withstand the OBE excitation. Analysis alone, without testing, is acceptable as a basis for qualification only if the necessary operability of the equipment is ensured by its structural integrity.

The applicant analyzed the complex equipment and demonstrated the structural integrity. This is sufficient for some of the equipment. However, for that equipment whose functionality is not ensured by structural integrity alone, the applicant has established a surveillance and testing program as indicated in the submittal of February 8, 1983. The submittal further stated that the electrical equipment had been purchased for most part to the requirements of IEEE Stds 323-1971 and 344-1971 and included the operability requirements in those standards. Moreover, the characteristics of the required response spectrum for Waterford 3, in general, are single peak and rather low magnitude. On the basis of these features and particularly of the commitment to a surveillance and testing program as outlined in concern (2), the staff concludes that adequate additional justification has been provided for Waterford 3 for the concern to be closed.

Status of Specific Concerns

All equipment-specific qualification concerns identified from the first and second audits have now been resolved. In the case of holdup Tank C (NSSS-PE-33) where modifications to the equipment were required, the completion of the modifications has been confirmed. In addition, the applicant has confirmed by submittal dated March 2, 1984, that all safety-related equipment is now qualified.

Conclusion

On the basis of the SQRT audit findings as well as the review of subsequent submittals, the staff concludes that an appropriate seismic and dynamic qualification program has been defined and implemented that provides adequate assurance that such equipment should function properly during and after the excitation vibratory forces imposed by the safe shutdown earthquake.

4 REACTOR

4.2 Fuel System Design

4.2.2 Design Evaluation

4.2.2.9 Seismic and LOCA Loadings

An important aspect of the behavior of the reactor core during a loss-of-coolant accident (LOCA) is the response of the fuel assemblies to asymmetric blowdown loads and the safe shutdown earthquake (SSE). The applicant has submitted by letter dated August 12, 1983, a plant-specific analysis described in the Combustion Engineering Inc. (CE) Topical Report CEN-159(C)-P, Revision 1-P, "Final Assessment of Waterford-3 Fuel Structural Integrity Under Faulted Conditions," dated July 15, 1983. The CEN-159(C)-P, Revision 1-P, results are based on the models and acceptance criteria described in the approved CE Topical Report CENPD-178, Revision 1.

Table 1 of CEN-159(C)-P, Revision 1-P, shows that the peak combined loads are well below the allowable prescribed limits in almost all cases except for the case of peripheral assemblies under a one-sided load. The peak combined load on these peripheral assemblies under a one-sided loading condition is about 300 lb higher than the allowable limit. The applicant performed an emergency core cooling system (ECCS) analysis to determine the coolability of these assemblies by assuming grid deformation according to the recommendations of SRP Section 4.2, Appendix A. The result shows that coolability is maintained mainly because these assemblies are located at a low power density area.

Therefore, because (1) the peak combined loads on the grids are below the prescribed limits for most cases and (2) the coolability is maintained for those fuel assemblies with combined load exceeding the prescribed limit, the staff concludes that the seismic and LOCA loading on fuel assemblies satisfies the intent of SRP Section 4.2, Appendix A, and this license condition can, therefore, be removed for Waterford 3.

4.4 Thermal-Hydraulic Design

4.4.1 Thermal-Hydraulic Design Criteria and Design Bases

In a letter dated March 1, 1984, the applicant provided a revised table of values for rod bow penalty that are slightly more conservative (0.5%) than the values given in Supplement 5 (Section 4.4.1). These values, which follow the guidelines of the CE Topical Report CENPD-225, an approved report, are given below:

Burnup (GWD/MTU)	Departure from nucleate boiling ratio penalty (%)
0-10.0	0.5
10.0-20.0	1.0
20.0-30.0	2.0
30.0-40.0	3.5
40.0-50.0	5.5

The applicant has stated that the thermal margin reductions for rod bow will be put into the departure from nucleate boiling ratio (DNBR) limit basis of the core protection calculator (CPC) system. They will be verified to be included in the DNBR limit calculations in the core operating limits supervisory system (COLSS) and the CPC system at least once every 31 days. Therefore, the appropriate provisions will be incorporated into the Technical Specifications. The applicant should also insert into the basis of the Technical Specifications any generic or plant-specific margin that may be used to offset the reduction in DNBR resulting from rod bowing and should reference the source and staff approval of each generic margin. With these requirements satisified by the applicant, the staff concludes that the applicant has adequately accommodated the reductions listed above.

The core flow design basis requires a minimum flow that will pass through the fuel region and be effective for fuel rod cooling as a percent of the primary coolant flow rate or 148.0×10^6 lb/hr. The remainder of the flow, called bypass flow, will be ineffective for cooling because it will take the following bypass paths:

(1) outlet nozzle/core support barrel (CSB) gap
(2) core shroud/CSB annulus
(3) alignment keys
(4) guide tubes

The design and best-estimate bypass flow rates for Waterford 3 were reduced in Amendment 30 of the FSAR from earlier values as shown below:

	Previous (%)	Current (%)
Design bypass flow	3.5	2.6
Calculated best-estimate bypass flow	2.7	2.1

The staff requested that the applicant provide a description and justification of changes resulting in the reduced bypass flow. This information was provided in a letter dated February 22, 1984, which indicated that two design changes were made to reduce the bypass flow through the guide tubes. The bypass flows in the other leakage paths remain the same. The best-estimate bypass flow rate in the guide tubes was reduced by (1) reducing the overall flow area of inlet flow holes and (2) adding sleeves in the upper ends of the guide tubes. These changes reduce the bypass flow by increasing the hydraulic resistance and are summarized below:

	Best-estimate bypass flow rate	
Bypass flow path	Previous (%)	Current (%)
Outlet nozzle/CSB gap	0.61	0.61
Core shroud/CSB annulus	0.62	0.62
Alignment keys	0.09	0.09
Guide tubes	1.38	0.78
Total	2.70	2.10

The applicant presented a description of the guide tube design changes and a breakdown of the bypass flow through the guide tubes, including the flow networks used to calculate the present best-estimate guide tube leakage rate of 2.1%. An additional 0.5% increment over the best-estimate value of 2.10% accounts for the effects of core crudding, tolerances, and other unknown factors and results in a design value of 2.6%. The staff concludes from the calculations presented that reduction from 3.5% to 2.6% for design bypass flow is acceptable.

4.4.2 Core Protection Calculator/Control Element Assembly Calculator

4.4.2.4 Verification of CPC/CEAC Software Modification Implementation

In Supplement 5 to the SER, the staff identified a deficiency in the Waterford 3 core protection calculator (CPC) software where a predetermined penalty factor for the both-failed control element assembly calculator (CEAC) condition was not applied for the local power density calculation. The staff required that the CPC software be corrected to conform to its design specification before the issuance of an operating license. The applicant in a letter dated October 20, 1983, notified NRC that the deficiency has been corrected in the LP&L, Revision 1, CPC software, which was generated in accordance with Combustion Engineering Topical Report CEN-39(A)-P, Revision 2, and that the Phase I and Phase II tests have been completed with satisfactory results and do not differ from those of the Revision 0 software tests. The staff, therefore, concludes that the Waterford 3 CPC/CEAC is acceptable.

5 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

5.2 Integrity of Reactor Coolant Pressure Boundary

5.2.4 Reactor Coolant Pressure Boundary Inservice Inspection and Testing

This section was prepared with the technical assistance of Department of Energy (DOE) contractors from the Battelle Pacific Northwest Laboratories and supplements the conclusions in Section 5.2.4 of the SER and Supplement 5, which addressed the definition of examination requirements and the evaluation of compliance with 10 CFR 50.55a(g).

The design of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Class 1 and 2 components of the reactor coolant pressure boundary (RCPB) incorporates provisions for access for inservice inspections, as required by Paragraph IWA-1500 of Section XI of the ASME Code. 10 CFR 50.55a(g) defines the detailed requirements for the preservice and inservice inspection (PSI and ISI) programs for light-water-cooled nuclear power facility components. On the basis of a construction permit date of November 14, 1974, this section of the regulations requires that a PSI program be developed and implemented to meet the requirements in Section XI of the ASME Code and addenda applied to construction of the particular components. Also, the initial ISI program must comply with the requirements of the latest edition and addenda of Section XI of the ASME Code in effect 12 months before the date of issuance of the operating license, subject to the limitations and modifications listed in 10 CFR 50.55a(b). In a letter dated February 9, 1983, the applicant submitted a PSI program for examinations that were conducted at the plant site based on the 1977 Edition of Section XI of the ASME Code including Addenda through Summer 1978. The visual inspection program is being conducted in accordance with the 1980 Edition of Section XI including Addenda through Winter 1980. The preservice examination of the welds in the principal components of the RCPB, such as the reactor pressure vessel (RPV), steam generators, pressurizer, and reactor coolant pump casings, was performed in the fabrication shop in a manner similar to that at other Combustion Engineering (CE) plants based on CE Document Number TR-ESS-037 entitled "Shop Preoperation Inspection Program." In Supplement 5, the staff determined that the PSI program submitted by the applicant on February 9, 1983, is acceptable on the basis of a review of the selection of welds subject to examination and the evaluation of the methods of volumetric examinations conducted at the plant site.

5.2.4.1 Evaluation of Compliance With 10 CFR 50.55a(g)

In a letter dated January 27, 1983, the applicant provided a detailed comparison of the preservice examination of the RPV performed in the CE fabrication shop with RG 1.150. The applicant also identified the near-surface areas that were electronically gated out, discussed the acoustical similarity of the welds, and described the physical limitations to volumetric examination. The preservice examination of the RPV was based on the 1974 Edition of Section XI of the ASME Code including Addenda through Summer 1974 and predates RG 1.150 by more than 5 years.

The inspections included the 0°, 45°, and 60° examinations of the beltline region and all circumferential and longitudinal welds. In addition, the nozzle-to-shell weld was examined from the nozzle bores in the perpendicular direction, and the flange-to-upper shell was examined from the closure flange mating surface. Electronic gating was used to eliminate the reflection from the cladding-water interface and the disturbance created by the cladding-parent metal interface at the front surface. The angle beam channels were electronically gated to include the far surface, and the straight beam gate was set as close as possible to the far-surface signal without continuously alarming the system. An estimate was provided of the near-surface gating for each RPV weld, which shows that for all of the 45° and 60° examinations, the gating varied from 1.12 to 2.0 in.

The core stabilizing lugs and the outlet nozzle knuckle are permanent physical obstructions preventing access to some portion of the welds subject to volumetric examination. Other interferences resulting in a liftoff of the transducer were caused by the vessel configuration or surface roughness. Liftoff produces a spurious indication on the recorder. The applicant has identified these limitations to examination in the tables and figures attached to the letter dated January 27, 1983. The staff's review of the applicant's letter describing the reactor pressure vessel examination has determined that the instrument performance checks, calibration, recording, and reporting of flaw indications were in accordance with the applicable provisions of Section XI of the ASME Code. The staff concludes that the electronic gating of the near-surface ultrasonic signal and the regions not examined because of physical limitations are consistent with the commercial practice at the time of the inspection. The staff has determined that the preservice examination based on Section XI and the radiography performed during construction provide an adequate level of preservice structural integrity.

In letters dated July 25, 1983, and February 10, 1984, the applicant requested relief from ASME Code requirements that he determined to be impractical and provided a supporting technical justification. The staff has determined that certain ASME Code, Section XI, examination requirements defined in 10 CFR 50.55a(g)(3) are impractical. Therefore, pursuant to 10 CFR 50.55(a)(2), the staff has allowed relief from the requirements that are impractical and that, if implemented, would result in hardships or unusual difficulties without a compensating increase in the level of quality and safety. On the basis of the granting of relief from these preservice examination requirements, the staff concludes that the PSI program for Waterford 3 is in compliance with 10 CFR 50.55a(g)(3). A detailed evaluation supporting this conclusion is provided in Appendix D to this report. The initial inservice inspection program for Unit 3 will be evaluated after the applicable ASME Code edition and addenda can be determined based on 10 CFR 50.55a(b) and before the first refueling outage when inservice inspections will be performed.

Periodic inspections and hydrostatic testing of pressure-retaining components of the reactor coolant pressure boundary in accordance with the requirements of Section XI of the ASME Code and 10 CFR 50 will provide reasonable assurance that evidence of structural degradation or loss of leaktight integrity occurring during service will be detected in time to permit corrective action before the safety functions of the components are compromised. Compliance with the inservice inspections required by Section XI of the ASME Code and 10 CFR 50 constitutes an acceptable basis for satisfying the inspection requirements of GDC 32.

5.4 Component and Subsystem Design

5.4.3 Shutdown Cooling (Residual Heat Removal) System

In Section 5.4.3 of Supplement 3, the staff indicated that it had not yet received the applicant's response to the requests for additional information with respect to the power-operated relief valve (PORV) issue. The supplement stated that if the responses were not provided before the anticipated fuel loading date for Waterford 3, the staff would require that the applicant provide a justification for safe operation of the plant in the interim.

In a letter dated September 20, 1983, the applicant submitted the above required responses.

The staff's evaluation of the need for a rapid depressurization capability for the current 3,410 MWt and 3,800 MWt classes of plants designed by Combustion Engineering (CE) consisted of reviewing the licensee, applicant, and vendor reponses to staff questions supplemented by independent analyses. The overall evaluation was grouped into four topic areas. First, the staff determined if the CE plants met current regulatory requirements without PORVs. Second, the staff determined the extent to which the existing design without PORVs can mitigate events that are beyond the design basis, and whether a PORV would substantially improve the ability of the plant to mitigate or reduce the severity of these events. Third, a probabilistic risk assessment (PRA) was performed to estimate the change in core melt probability if a PORV were installed. And fourth, the cost and benefits were assessed and compared.

The results of the staff review led to the conclusion that, on the basis of risk reduction and cost/benefit consideration, there was no overwhelming benefit that would be obtained by requiring the installation of PORVs in CE plants that currently do not have them. However, when other considerations regarding the potential benefit of a PORV are factored into the evaluation, more substantial benefits can be realized. Given the high degree of uncertainty in the evaluation results, along with the qualitative nature of the basis on which any decision would have to be made, the staff concludes that the decision regarding PORVs for these CE plants should be deferred and incorporated into the technical resolution of Unresolved Safety Issue (USI) A-45. Because part of the benefit of the PORVs was predicated on their ability to provide an alternate decay heat removal path (feed and bleed), any improvements in decay heat removal capability that might be promulgated as a result of the A-45 assessment could reduce the net benefit of PORVs. Finally, the events for which PORVs could prove to be a benefit are of low probability, and the staff is aware of no immediate safety concerns associated with deferring the PORV decision until the A-45 decision is made. The staff's detailed evaluation is documented in draft NUREG-1044, dated March 1984.

The staff briefed the Advisory Committee on Reactor Safeguards (ACRS) Subcommittee on Decay Heat Removal Requirements (on October 4, 1983) and the full ACRS in an executive session (on October 13-15, 1983) on its evaluation. Subsequently, the ACRS issued a letter dated October 18, 1983, which stated that the Committee agrees with the NRC staff's recommendation to integrate any new requirements for rapid depressurization into the more comprehensive new requirements for improvements to decay heat removal systems expected to be forthcoming from Task Action Plan A-45 within 1 year. The committee saw no need for earlier resolution of the PORV issue. A copy of this letter is included in Appendix C to this supplement.

The auxiliary pressurizer spray (APS) in CE plants without PORVs is used for the reactor coolant system depressurization function to achieve cold shutdown in accordance with Branch Technical Position (BTP) RSB 5-1 (NUREG-0800). Position A.1 of BTP RSB 5-1 states that the reactor should be capable of being brought from normal operating conditions to cold shutdown with safety-related systems. However, in accordance with the recommended implementation of BTP RSB 5-1 for Class 2 plants (Waterford 3 is a Class 2 plant), the APS design need not be classified as safety related (i.e., single failure proof) if (1) manual actions inside containment after a safe shutdown earthquake or single failure or (2) remaining at hot standby until manual actions or repairs are completed is found to be acceptable for the individual plant. No information has been submitted to show conformance with either of these positions.

Also, the Waterford plant uses APS for plant depressurization following a steam generator tube rupture (SGTR) accident. It has been general practice in staff casework reviews to require that the systems and components relied on for mitigating design-basis accidents and ensuring that the radiological consequences are within 10 CFR 100 guideline values be classified as safety related and, therefore, designed to safety-grade standards, which include meeting the single-failure criterion.

The applicant has been requested to provide sufficient justification to demonstrate that the Waterford APS design meets the criteria of BTP RSB 5-1 for Class 2 plants and the criteria for systems required for SGTR accident mitigation. On February 29, 1984, the applicant submitted a report (CE Topical Report CEN-259) to address the BTP RSB 5-1 requirement with regard to natural circulation cooldown and depressurization. The report asserted that Waterford 3 could cool down to RHR initiation conditions without APS operation and meet the BTP RSB 5-1 positions. This report is currently under staff review.

The staff's review of the APS system is still considered open and the staff will report its evaluation in a future supplement to the SER.

In Section 5.4.3 of the SER, the staff stated that "subsequent to the SONGS tests, but prior to fuel loading at Waterford, the staff will require that the applicant submit a review of the SONGS tests and demonstrate the acceptability and applicability of the results to the Waterford 3 plant." In response to the above staff request, the applicant in a letter dated February 29, 1984, submitted a report (CE Topical Report CEN-259) to address the acceptability and applicability of the SONGS test results to the Waterford 3 plant. This CE topical report is currently under staff review and the staff will report its evaluation of this report in a future supplement to the SER. A license condition is not needed for this issue.

6 ENGINEERED SAFETY FEATURES

6.2 Containment Systems

6.2.4 Containment Isolation System

In a letter dated February 16, 1984, the applicant proposed a design modification to the Waterford 3 component cooling water (CCW) system. The objective of the design change is to ensure the uninterrupted supply of cooling water from the CCW system to the reactor coolant pump (RCP) seal coolers in the event a safety injection actuation signal (SIAS) or containment isolation actuation signal (CIAS) is generated by low pressurizer pressure. As discussed in the proposed change, a flow path will be provided to the RCP seal coolers using the CCW A train, by delaying closure of the header isolation valves and three containment isolation valves. This will be accomplished by changing the isolation signals from SIAS (header isolation valves) and CIAS (containment isolation valves) to the containment spray actuation signal (CSAS). The SIAS and CIAS occur on either high containment pressure or low pressurizer pressure; CSAS occurs on high containment pressure and is interlocked with SIAS. The proposed change will only affect the isolation of CCW for those transient events that result in low pressurizer pressure without a concomitant high containment pressure condition. Upon CSAS, the flow pattern and containment isolation of the CCW A train will be exactly as it is now after SIAS or CIAS.

Repeated interruption of cooling water to the RCP seal coolers, resulting from certain transient events or inadvertent SIAS and CIAS, would increase the likelihood of seal degradation and could eventually cause the seals to leak. On the other hand RCP operation would help mitigate certain adverse effects that could result from certain abnormal transient events and many accident sequences. The NRC staff discussed the issue with the applicant during a meeting held on January 26, 1984, and concurred with the applicant that overall plant safety would be improved by maintaining RCP operation for a wider range of transient events, even though the diverse isolation criteria for the A train of the CCW system are compromised.

The diverse isolation signal criteria normally must be met for all nonessential systems that penetrate the containment. Exception may only be taken if a need can be established for use of the system during certain accident sequences or transient events. As discussed above and indicated in the applicant's submittal, continuous RCP operation would improve plant operational safety, and thus, the CCW system is no longer considered a nonessential system. The affected containment penetration would continue to be isolated on high containment pressure if a LOCA were detected in the containment, and the proposed changes do not affect the FSAR Chapter 15 safety analysis and do not pose an unreviewed safety issue. Consequently, the staff finds acceptable the proposed changes as they relate to the containment isolation system design.

In the SER, an outstanding issue was identified concerning the minimum containment setpoint pressure, that is, Position 5 of TMI Item II.E.4.2. This was necessary because the FSAR indicated that the required study to determine the setpoint pressure was not completed.

By letter dated March 12, 1984, the applicant proposed a 17.1-psia setpoint pressure. However, because the total instrument error contained in the setpoint pressure is greater than 1.0 psi, the staff requested detailed justification for the instrument error. The applicant provided the justification by letter dated April 19, 1984. The staff has reviewed these submittals and concurs with the applicant's justification for the proposed setpoint pressure. Therefore, the staff concludes that the outstanding issue on containment setpoint pressure is satisfactorily resolved.

6.2.4.1 Demonstration of Containment Purge and Vent Valve Operability

Demonstration of operability of the containment purge and vent valves, particularly the ability of these valves to close during a design-basis accident, is necessary to ensure containment isolation. This demonstration of operability is required by BTP CSB 6-4 and SRP Section 3.10 for containment purge and vent valves which are not sealed closed during operating conditions 1, 2, 3, and 4.

The Waterford 3 purge and vent system containment isolation valves are Fisher Control 40-in. 9220 series butterfly valves equipped with Bettis pneumatic actuators. The actuators are air open-spring close type model T420-SRI-M3. The valve tag numbers are 2HV-13150B, B151A, B152A, B153B, B154B, and B155A.

The applicant has provided operability demonstration information for the purge and vent system isolation valves of Waterford 3 in submittals dated August 19, 1981, October 16, 1981, June 25, 1982, and September 16, 1983.

The applicant's original submittal demonstrating operability of the purge and vent valves was not based on any specific loss-of-coolant-accident (LOCA) case. A 44-psig pressure was given as the potential peak containment pressure with the assumption made that this was the differential pressure across the valve at all disc angles during closure. The applicant's review determined that the limiting angle to which the valve may be opened was 40°. The staff's evaluation of the submittal identified four open times to which the applicant was requested to respond. The response to these items in a letter dated September 16, 1983, resulted in a reanalysis of the highly stressed components and the containment-pressure-versus-valve-position relationship. This reanalysis permitted a maximum valve open position of 52°. The applicant in submitting responses to the open items thus stated that 52° is now the limiting angle. Subsequent conversations with the applicant and EBASCO staff confirmed operability of the purge and vent valves from the 52° open position.

The design-basis-accident (DBA)/LOCA condition simulated was a double-ended hot-leg break (DEHLB) totaling 19.24 ft^2. A containment response-time/pressure chart for this accident was provided. This chart shows an elapsed time of 1.4 sec from receipt of the solenoid vent signal until the purge valves begin to close and 6.4 sec from the initiation of the accident until the valves are fully closed. Closing-time data were obtained by testing one of the installed valves. A conservative assumption is made by the applicant that the closure time from the partially open position will require as much time as closure from the 90° position. This is because substantial time is required to bleed off or vent the actuator piston pressure to the point where the spring can overcome the pressurization force and begin to close the valve. If each of the six valves is installed with similar connnections between the solenoid and actuator and if supply pressure is the same, similar closure times can be expected.

The applicant did not provide drawings or sketches showing the pipe/ducting configuration of valve installation information. The applicant, however, did state that all Fisher sizing data are based on dynamic torque determination tests that were performed with uniform profiles and on valve discs with representative geometries. Upstream of the Waterford purge valves, there is only a straight run of duct with no elbows, T-connections etc. Flow through the valve is expected to be uniform. This piping configuration description was confirmed during the original NRC evaluation. It was indicated that a tee existed (inside containment) on the exhaust line. However, a damper installed in the tee'd leg is kept closed and is opened only during refueling.

The applicant submitted an Instrument Society of America (1969) paper that presented a technique developed by Fisher by which butterfly valve shaft torque (resulting from fluid flow) can be determined for both compressible and incompressible flow. This paper is meant to generally describe and justify Fisher's method of determining dynamic torques for its line of butterfly valves. It should not be interpreted as a test report concerning the tests on actual model valves representative of the Waterford valves.

The shaft stresses caused by seismic loading of the valve disc were determined using an ANSYS finite-element model. The ANSYS analysis assumed worst-case, horizontal-shaft orientation. Valve orientation does not have a strong effect on total shaft stresses as a result of the dominance of dynamic flow effects on the disc. The ANSYS finite-element program determined stresses in the valve shaft resulting from a resultant seismic load of 1.4 g. The shaft program was used to calculate shaft stresses resulting from dynamic torque and pressure drop at various angles of opening. These stresses represent the effects of dynamic torque and severe conditions.

The applicant submitted a computer printout that represents a calculation of the six critical stress values in the valve shaft at various angles of closure accounting for the pressure conditions at that time. These stresses were then combined manually with the seismic stresses determined from the ANSYS program. The combined principal stresses at particular opening angles were compared with the allowable values in bending and shearing to determine an acceptable opening angle using the 0.75S shear stress allowable. The maximum opening angle was determined to be 52° using interpolated stress valves.

In response to the NRC concern as to whether the allowable shear stresses should be 0.75S or 0.6S, the applicant presented the following discussion in the letter dated September 16, 1983, which the staff finds acceptable.

> Use of the premium strength 17-4PH material for the shaft justified 0.75S as the shear stress allowable, since the highest torsional stresses experienced occur only in the outermost layer of the shaft

material. Paragraph NB-3227.2 of Section III of the ASME Boiler and Pressure Vessel Code recognizes this distinction, stating that 0.8Sm is suitable as the allowable for shafts in torsion. When applied to the stress value "S" found in Section VIII, the allowable is less than yield.

Table I.7.1, Appendix I of Section III, lists S of 35,000 psi for shafts manufactured from 17-4PH hardened to H1100, so 0.75S = 26,250 psi. Minimum tensile yield is given as 115,000 psi, so minimum yield in shear would be at least 57,500 psi, providing a substantial margin.

A torque capacity chart was provided for the actuators used on the purge and vent valves. The applicant's submittal indicates that the actuator torque output drops from 100% of the end of stroke torque at 0° (48,500 in.-lb) to 76% of this value when the valve is 50° open. The torque capacity chart presented assumed a constant ΔP of 39 psi during the entire valve closure time. The chart demonstrates that the actuator will close the valve for all open angles.

The 52° maximum valve open position will be limited by mechanical stops. The applicant's surveillance requirements will verify that the valves open to less than or equal to 52° every 18 months and/or following any adjustment of the mechanical position stops.

For the valves inside containment, the applicant has considered and addressed the containment pressure rise, that is, the effect of this back pressure on valve operability. The Bettis actuators are designed with a vent port (open to ambient) on the spring side of the piston. A three-way solenoid valve is used to vent the other side of the piston. The applicant believes that this design is not affected by the back pressure.

The actuator design described precludes the existence of differential pressure across the piston resulting from containment pressure. The venting rate from the opening side of the piston and the torque margin available from the actuator are not affected by the back pressure to the extent that valve stroke time is increased under the accident condition compared with the no-load stroke time.

The staff has completed its review of information submitted to date concerning the operability of the containment purge and vent valves for Waterford 3. It finds that the information submitted has satisfactorily demonstrated the ability of the containment purge and vent valves, when limited to 52° or less, to close against the buildup of containment pressure in the event of a DBA/LOCA.

6.6 Inservice Inspection of Class 2 and 3 Components

This section was prepared with the technical assistance of DOE contractors from the Battelle Pacific Northwest Laboratories and supplements the conclusions in Section 6.6 of the SER and Supplement 5, which addressed the definition of examination requirements and the evaluation of compliance with 10 CFR 50.55a(g).

10 CFR 50.55a(g) defines the detailed requirements for the preservice and inservice inspection (PSI and ISI) programs for light-water-cooled nuclear power facility components. On the basis of a construction permit date of

November 14, 1974, this section of the regulations requires that a preservice inspection program be developed and implemented to meet the requirements in Section XI of the ASME Code and Addenda applied to construction of the particular components. Also, the initial inservice inspection program must comply with the requirements of the latest edition and addenda of Section XI of the ASME Code in effect 12 months before the date of issuance of the operating license, subject to the limitations and modifications listed in 10 CFR 50.55a(b). In a letter dated February 9, 1983, the applicant submitted a preservice inspection (PSI) program for examinations that were conducted at the plant site based on the 1977 Edition of Section XI of the ASME Code including Addenda through Summer 1978. The visual inspection program is being conducted in accordance with the 1980 Edition of Section XI including Addenda through Winter 1980. The preservice examination of the welds in the steam generators was performed in the fabrication shop in a manner similar to that at other Combustion Engineering (CE) plants based on CE Document Number TR-ESS-037 entitled "Shop Preoperation Inspection Program." In Supplement 5, the staff determined that the PSI program submitted by the applicant on February 9, 1983, is acceptable on the basis of a review of the selection of welds subject to examination and the evaluation of the methods of volumetric examinations conducted at the plant site.

In letters dated July 25, 1983, and February 10, 1984, the applicant requested relief from ASME Code requirements that he determined to be impractical and provided a supporting technical justification. The staff has determined that certain ASME Code, Section XI, examination requirements defined in 10 CFR 50.55a(g)(3) are impractical. Therefore, pursuant to 10 CFR 50.55a(a)(2), the staff has allowed relief from the requirements that are impractical and that, if implemented, would result in hardships or unusual difficulties without a compensating increase in the level of quality and safety.

On the basis of the granting of relief from these preservice examination requirements, the staff concludes that the PSI program for Waterford 3 is in compliance with 10 CFR 50.55a(g)(3). A detailed evaluation supporting this conclusion is provided in Appendix D to this report. The initial ISI program for Unit 3 will be evaluated after the applicable ASME Code edition and addenda can be determined based on 10 CFR 50.55a(b) and before the first refueling outage when inservice inspections will be performed.

Compliance with the inservice inspections required by Section XI of the ASME Code and 10 CFR 50 constitutes an acceptable basis for satisfying the applicable inspection requirements of GDC 36, 39, 42, and 45.

7 INSTRUMENTATION AND CONTROLS

7.1 Introduction

7.1.2 Specific Findings - Open Items

(2) IE Bulletin No. 79-27 (Capability for Safe Shutdown Following Loss of a Bus Supplying Power to Instruments and Controls)

It is stated in the Waterford 3 SER (NUREG-0787) that the applicant should provide for staff review information related to IE Bulletin 79-27 guidelines. The applicant responded by letters dated February 27 and May 7, 1984. Review of the information and discussions with the applicant revealed that all Class 1E and non-Class 1E ac and dc buses supplying power to safety-related and nonsafety-related instrumentation and control systems that could affect the ability to achieve a cold shutdown condition were considered using the guidelines of IE Bulletin 79-27. The study included a component failure mode and effects analysis for each individual load on each bus. The applicant stated that no case was found where design modifications were required to ensure the ability to achieve a cold shutdown condition.

The information provided addressed the various IE Bulletin 79-27 concerns. The applicant informed the staff that all the Waterford 3 instrumentation and controls required for safe plant shutdown are redundant and Class 1E. On the basis of this capability, the control room operators have the necessary redundant Class 1E instrumentation and control systems available to obtain cold shutdown. The applicant has stated that appropriate annunciation is provided in the control room which will indicate the loss of a particular bus. The applicant has prepared procedural guidelines to address adverse effects from single instrument bus losses. These procedures were developed to ensure that the capability exists to achieve cold shutdown if power is lost to any one Class 1E or non-Class 1E instrument bus. Also, during rereview of IE Circular 79-02, which included both Class 1E and non-Class 1E inverter supplied instrumentation and control buses, the applicant concluded that no modifications were necessary.

The staff has concluded that satisfactory information (as summarized above) has been provided to address the IE Bulletin 79-27 concerns. Therefore, this issue is considered to be resolved.

7.3 Engineered Safety Features Actuation System

7.3.4 Emergency Feedwater System

It is stated in Supplement 1 that the applicant should provide information to (1) confirm compliance of the emergency feedwater (EFW) valve control system with the design criteria applicable to the plant protective system, (2) describe the effects on the EFW system reliability analysis as a result of the

EFW valve control scheme addition, and (3) confirm the acceptability of the interface between the emergency feedwater actuation signal (EFAS) and the main steam isolation signal (MSIS). It should be noted that the EFW valve control system was accepted by the staff on a functional basis but required further confirmation through the submission and review of final design drawings.

The protective function requirements associated with the EFW system are

(1) the capability to supply emergency feedwater to the intact steam generator when required following a pipe break

(2) the capability to control steam generator level (flow control)

(3) the capability to isolate the EFW system from a faulted steam generator as required

The staff reviewed the Waterford 3 design related to the emergency feedwater actuation signal initiation including the feed-only good steam generator logic and found this portion of the design acceptable with the exception of a confirmatory issue associated with the function of the MSIS. The EFW valve control system used for automatic level control and the EFAS-MSIS interface are evaluated further below.

The EFW valve control system interfaces with the EFAS and the feed-only good steam generator logic, which includes use of the main steam isolation signal. FSAR Section 7.3.1.1.6 provides a description of the EFW system and associated interfaces.

During review of the EFW valve control system, the staff focused attention on the EFW system functional requirements as described above as part of background information. The purpose of this control system is to regulate emergency feedwater flow to the steam generators so as to minimize adverse effects on the reactor coolant system (overcooling condition, etc.). In the automatic mode, the control valves (four total - two per steam generator) are positioned by signals derived from the emergency feedwater flow and steam generator wide-range level measurement instrumentation loops. The EFW valve control system is designed as a safety-related system.

Several issues were identified during staff review which required resolution. During the initial stage of the review, the staff revealed that a single failure of a steam generator level transmitter (fail high) could inhibit the automatic initiation of EFW flow to an intact steam generator. This was possible because of the shutoff and control valve arrangements shown in FSAR Figure 7.3-13. There are four valves per steam generator - two train A valves and two train B valves with each train powered from redundant power supplies and consisting of a shutoff and a control valve. The train A and B shutoff and control valves were controlled by a single train A and B level transmitter, respectively, for a particular steam generator. The postulated event was corrected by removing the steam generator level control signal contacts from the train A and train B shutoff valve circuits associated with each steam generator. Thus, the shutoff valves will be opened by an EFAS only and will not be released to control by the level transmitter. Drawings were submitted and reviewed by the staff. This modification negates the above described single-failure concern and is, therefore, acceptable. It should be noted that with this design modification a failure of any one level transmitter will disable only one train-associated EFW control valve.

With the EFW valve control scheme in the automatic mode, the operator depends on the system to automatically control the steam generator level. The staff expressed a concern associated with the development of a potential steam generator overfill condition after a single failure is assumed in the automatic circuits after initiation of EFW. This situation is compounded by the fact that the manual initiation capabilities of the shutoff valves are defeated on generation of an EFAS as long as the steam generator level is less than 74%. The applicant has verified that transient analyses (performed by Combustion Engineering (CE)), which result in EFW system initiation, show that a low steam generator pressure signal would be generated before a possible overfill condition is reached should a single failure in the level control circuits allow the steam generator to continue to fill. Initiation of this signal generates an auto-reset signal (discussed below) and an MSIS which, in combination, will give a priority-close command to automatically close both the shutoff and control valves which are associated with opposite trains within each flow path. The MSIS will also cause closure of the MSIVs. Discussions with the applicant revealed that the subject CE analyses are available for audit at the site should it be necessary. It should be noted that as a backup to this automatic feature (priority-close), the system allows the operator to reset EFAS when the steam generator level reaches 74%. Upon reset of EFAS at this point, the operator would have approximately 1 hour (with maximum emergency feedwater flow) to manually close the shutoff valves through the operation of control switches provided in the control room before water starts entering the main steam lines. Also, the applicant has confirmed that worst-case effects caused by overfilling to the main steam isolation valves during all modes of plant operation were analyzed in the FSAR and were found to be acceptable. Further verification of such overfilling effects has been provided through preoperational hydrotests. On the basis of the above discussion, the staff considers this issue resolved.

As stated above, the EFAS is reset automatically under certain conditions. This auto-reset feature occurs on a low steam generator pressure signal which also initiates an MSIS. The EFW pumps are not affected by this reset. The reset of EFAS on low steam generator pressure in combination with MSIS will generate a priority-close signal as part of the feed-only good steam generator logic. This signal is used to quickly isolate the steam generator (close all associated EFW control and shutoff valves) in order to determine which steam generator is intact. Upon this determination, the priority-close signal is deactivated automatically (EFAS reinstated to the intact steam generator) upon a valid differential pressure signal coincident with a low steam generator level signal. The applicant incorporated this feature into the design since operator action may not be adequate to execute such a function in a timely manner because of the sequence of events that may result from a postulated steamline rupture. Also, the applicant analyzed the operating modes of the priority-close signal and its associated single failures. It was concluded that no single failure would preclude the EFW system from performing the required function of supplying emergency feedwater flow to the intact steam generator.

Testing of the EFW valve control system logic was discussed. The applicant committed to perform appropriate functional tests on the channels associated with this level control system every refueling and upon obtaining a cold shutdown condition if not performed within a specified frequency. The Waterford 3 Technical Specifications are currently being prepared for issuance by the staff. Appropriate Technical Specification requirements will be implemented before final issuance. The staff considers this issue resolved.

An issue was pursued by the staff related to the failure within a single auxiliary relay cabinet (ARC) and its effect on the ability of the EFW system to perform the required protective functions. One set of shutoff and control valves is indirectly controlled by one train and its associated ARC while the other set (opposite train) of valves is controlled by its redundant counterpart. The train-associated shutoff and control valves are operated by separate actuation relays within a specific ARC. The applicant has performed a failure analysis and has concluded that there is no single failure within the ARCs which could disable both flow paths to the intact steam generator. The staff discussed the results of this analysis with the applicant and concurred with the findings. Therefore, this issue is considered resolved.

A reliability study of the EFW system was performed by the applicant and submitted as Appendix 10.4.9B to the FSAR (Amendment 13). Subsequent to this, the applicant evaluated the EFW system reliability based on the EFW valve control scheme. The applicant determined that the changes required by the addition of the subject control system had an insignificant effect on the original EFW system reliability analysis and that no revisions were, therefore, necessary. In fact, it was concluded by the applicant that the capability of the EFW system to respond properly to an event has been greatly enhanced. The new control scheme logic allows the system to respond only to the extent necessary for the specific transient through control valve modulation. Thus, a relatively minor transient will not be exacerbated into a more severe transient by an EFW overresponse (full flow) which could lead to increased engineered safety features systems and operator action challenges.

The staff concludes that the EFW valve control scheme enhances the overall plant safety and operability. The addition of the subject design appears to be an aid to the performance of the protective functions required and appears to be consistent with the design criteria applicable to the emergency feedwater system.

In conclusion, the staff considers the EFW system (including the associated valve control scheme) to be acceptable and adequately confirmed by the applicant for use as part of the Waterford 3 design.

7.4 Systems Required for Safe Shutdown

7.4.3 Emergency Shutdown From Outside the Control Room

Supplement 3 noted a staff concern related to the negation of automatic actuation of engineered safety features (ESF) functions. The applicant responded through discussions that the Waterford 3 design does not fully comply with the staff's position that transfer of control to the remote shutdown location should not disable any automatic actuation of ESF functions while the plant is attaining or is maintained in hot shutdown other than where ESF features are manually placed in service to achieve or maintain hot shutdown. The applicant committed to either modify the design to meet the staff's position or provide additional information to justify the existing design.

Subsequent to the above commitment, the applicant provided additional information by letter dated February 11, 1983. The information provided shows that the transfer of ESF functions to the remote shutdown panel (LCP-43), upon control room evacuation, will not defeat the automatic ESF response capability with the exception of the boric acid pumps. The boric acid pumps can be manually controlled from the LCP-43. The staff finds this acceptable.

On the basis of the above discussion and that in Supplement 3 (Section 7.4.3), the staff concludes that the Waterford 3 design adequately complies with the requirements of GDC 19 regarding remote safe shutdown from outside the control room.

7.5 Safety-Related Display Instrumentation

7.5.2 Postaccident Monitoring Instrumentation

The SER required the applicant to commit to meeting the intent of the prescriptive requirements of RG 1.97, which provides detailed guidelines for the instrumentation needed to monitor plant variables and systems during and following an accident in a light-water-cooled nuclear power plant. By letter dated July 6, 1983, the applicant responded to the requirements of RG 1.97 by outlining Waterford's capability to monitor the required postaccident parameters. Hence, the stipulation to meet the intent of RG 1.97 has been completed, and this issue has been satisfactorily resolved. A condition to the license is no longer necessary.

7.5.4 Safety Parameter Display System

In response to the guidelines of NUREG-0696, "Functional Criteria for Emergency Response Facilities," and as superseded by NUREG-0737, Supplement 1, the applicant has committed to provide for Waterford 3 a plant safety parameter display system (SPDS) and has provided a schedule for implementation of the SPDS.

NUREG-0737, Supplement 1, requires that the SPDS provide the operators with sufficient information on reactivity control, reactor core cooling and heat removal from the primary system, reactor coolant system integrity, radioactivity control, and containment conditions. This information is to aid the operators in determining the safety status of the plant.

In accordance with NUREG-0737, Supplement 1, an applicant must prepare a written safety analysis describing the basis on which the selected parameters are sufficient to assess the safety status of each identified function for a wide range of events, which includes symptoms of severe accidents. Also, the applicant's proposed implementation of an SPDS system and the Technical Specifications must be reviewed to determine whether the changes involve an unreviewed safety question or a change of Technical Specifications.

The staff's evaluation of the applicant's response to Generic Letter 82-33 (April 15, 1983) on emergency response capabilities will be presented in Section 22 of a subsequent supplement to the SER.

9 AUXILIARY SYSTEMS

9.1 Fuel Storage Facility

9.1.4 Fuel Handling System

As a result of Generic Task A-36, "Control of Heavy Loads Near Spent Fuel," NUREG-0612, "Control of Heavy Loads at Nuclear Plants" was developed. Following the issuance of NUREG-0612, a generic letter, dated December 22, 1980, was sent to all operating plants, applicants for operating licenses, and holders of construction permits requesting that responses be prepared to indicate the degree of compliance with the guidelines of NUREG-0612. The responses were to be made in two stages. The first response (Phase I, Section 5.1.1 of NUREG-0612) was to identify the load-handling equipment within the scope of NUREG-0612 and describe the associated general load-handling operations such as load paths; procedures; operator training; special and general purpose lifting devices; maintenance, testing, and repair of equipment; and handling-of-equipment specifications. The second response (Phase II) was intended to show that either single-failure-proof handling equipment was not needed or that single-failure-proof equipment had been provided. This supplement contains the staff's evaluation of Phase I. An evaluation of Phase II will be the subject of a future supplement.

By letter dated December 22, 1980, the applicant was requested to review the provisions for the handling and control of heavy loads at Waterford 3 to determine the extent to which the guidelines of NUREG-0612 are satisfied and to commit to mutually agreeable changes and modifications that would be required to fully satisfy these guidelines.

The staff and its consultant, Idaho National Engineering Laboratory (INEL), have reviewed the applicant's submittals for Waterford 3. As a result of its review, INEL has issued the technical evaluation report (TER) shown in Appendix I of this supplement. The staff has reviewed the TER and concurs with its findings that the guidelines of NUREG-0612, Sections 5.1.1 and 5.3, have been satisfied. Consequently, the TER is incorporated as a part of this supplement. The staff concludes that Phase I of NUREG-0612 for Waterford 3 is acceptable. The staff review of Phase II of NUREG-0612 for Waterford 3 will be the subject of a future evaluation. Until that review is complete, the following condition shall be included in the Waterford 3 operating license:

> Prior to startup following the first refueling outage, the applicant shall have made commitments acceptable to the NRC regarding the guidelines of Sections 5.1.2 through 5.1.6 of NUREG-0612 (Phase II: 9-month responses to the NRC generic letter dated December 22, 1980).

9.5 Other Auxiliary Systems

9.5.1 Fire Protection

9.5.1.4 Fire Protection of Safe Shutdown Capability

As a result of findings from an Appendix R audit conducted April 9-13, 1984, at the Waterford 3 site, the issue of alternative shutdown from outside the control room in the event of loss of offsite power and control room or cable vault burn-out is still considered open. The staff's evaluation of Louisiana Power & Light Company's response to this issue will be presented in a subsequent supplement to the SER.

13 CONDUCT OF OPERATIONS

Since the publication of Supplement 5 in June 1983, some organizational changes have been made or proposed by the applicant in later amendments to the FSAR. The staff has reviewed these changes through FSAR Amendment 34 dated January 13, 1984. This report provides the staff's evaluation of these changes.

13.1 Organizational Structure and Qualifications

13.1.1 Management and Technical Support Organization

The position of Vice President-Nuclear Operations no longer exists. The support groups and plant operations personnel that had reported to the Vice President-Nuclear Operations now report directly to the Senior Vice President-Nuclear Operations who reports to the President/Chief Executive Officer. The Senior Vice President-Nuclear Operations now has no responsibility for any corporate functions other than nuclear.

The staffing levels for Nuclear Operations are shown in Table 13.1.

The major support functions still exist, but some have been split among more than one organizational position. With reference to Figure 13.1, the technical functions of the former Nuclear Project Support Group are now spread among the Nuclear Services, Project Manager, and Completion Manager Groups.

The Nuclear Services Group (Figure 13.2) provides assistance in the areas of emergency planning, records and offsite clerical support, licensing and technical support, and special projects.

The Project Management Group (Figure 13.3) provides assistance in the areas of construction management, engineering, nuclear safety, contracts, costs, scheduling, and records management.

The Completion Manager is responsible for the transition of Waterford 3 from the construction phase to the operations phase. This responsibility includes ensuring that the initial testing and startup activities result in an effective transfer of plant systems to the plant staff. The Completion Manager functionally reports to the Project Manager for system transfers and reports to the Senior Vice President-Nuclear Operations in all other matters.

The Change Manager controls changes to the project that affect the budget, schedule, or technical baselines.

Although the training of plant staff is now the responsibility of the Plant Manager (see Section 13.2 of this supplement), the Training Evaluation and Assurance Group functions to provide assurance that the content and quality of the Nuclear Operations Department training programs meet stated goals in the areas of compliance with regulatory and industry standards.

The Safety Review Committee, Quality Assurance, and Plant Organization staff still report to the Senior Vice President-Nuclear Operations.

In addition to the support provided to Waterford 3 under the Senior Vice President-Nuclear Operations, other departments within LP&L will provide support, especially in the areas of offsite power transmission and offsite power interface with the facility, emergency planning and public information, personnel selection and recruiting, environmental affairs, procurement, and security.

A subsidiary of Middle South Utilities, Middle South Services, Inc. (MSS), provides specific services, usually of a specialized or technical nature, to Middle South Utilities and the system operating companies. These services encompass such areas as computer operations, engineering, reactor analysis, nuclear fuel management, quality assurance, and system operations coordination.

The staff believes that the applicant's reorganization of the technical and administrative functions for support of plant operations has produced a logical arrangement of these functions, which will provide definite lines of management control and adequate independence of those functions, such as quality assurance and safety reviews, that should be separate from the pressures of plant operations.

There have been some personnel changes in the key corporate positions. FSAR Amendment 34 includes résumés of the Senior Vice President-Nuclear Operations, the Corporate Quality Assurance Manager, the Nuclear Services Manager, the Plant Manager, the Project Manager, the Completion Manager, and the Change Manager. However, these résumés were so brief that the staff was unable to evaluate these individuals adequately to support a finding that the applicant complies with the requirements of 10 CFR 50.40(b). At the staff's request, the applicant provided, in a letter dated April 30, 1984, additional and more detailed résumés. The staff has reviewed these résumés and provides a summary of the experience of key technical support personnel below.

Although the Senior Vice President-Nuclear Operations has been with LP&L for only about 1 year, he brings to the Waterford project 27 years of increasingly responsible experience with the U.S. Navy, including 20 years in the naval nuclear power program. Following naval service, he served in technical management positions in the operation of two commercial nuclear power plants and in the engineering and construction of a third commercial nuclear power plant.

The Nuclear Services Manager has been with LP&L since 1975, following 8 years in the naval nuclear-powered submarine service. After a year at one of LP&L's fossil-fueled stations, he has been continuously involved with technical support of the Waterford project. His principal technical staff subordinates, most of whom have been with LP&L for a year or less, provide extensive commercial nuclear power plant experience in mechanical, nuclear, and metallurgical engineering.

The Project Manager has 26 years of construction and management experience with the U.S. Army Corps of Engineers. Before assuming his present position with LP&L in mid-1983, he held technical positions of responsibility in the engineering and construction of another commercial nuclear power plant. His principal technical staff subordinates, who have been with LP&L for about 3 to 20 years, have broad commercial nuclear power plant experience in mechanical end electrical engineering, nuclear safety and licensing, and instruments and controls.

On the basis of the staff's review of the changes to the corporate organization, its functions for supporting plant operation, and the staffing level for the Nuclear Operations organization, the staff concludes that the applicant has an acceptable organization and adequate resources to provide technical support for the operation of the facility.

13.1.3 Plant Organization

The present plant organization is shown in Figure 13.4. All the previous functions of the plant staff have been retained; only the organization of these functions has been modified.

By letter dated April 26, 1984, the applicant provided a shift organization chart, which is reproduced in this report as Figure 13.5. The shift supervisors report to the Operations Superintendent as shown in Figure 13.4. Besides the usual licensed and unlicensed operators, each shift will include a health physics technician, a rad-chem technician, and a shift technical advisor. Although not a regulatory requirement, the applicant expects each shift to include a third nuclear plant operator, two additional nuclear auxiliary operators, and two nuclear auxiliary operators in training. These will serve on the fire brigade. The auxiliary operators in training spend their time, when not assisting the auxiliary operators, working on becoming qualified to become reactor operators.

Plant staffing levels are presented in Table 13.1.

There have been significant changes in the personnel who fill key positions in the organization. FSAR Amendment 34 included résumés of most of the supervisory and management personnel. However, these résumés were so brief that the staff was unable to evaluate these individuals adequately to support a finding that the applicant complies with the requirements of 10 CFR 50.40(b). At the staff's request, the applicant provided, in letters dated April 30 and May 14, 1984, additional and more detailed résumés. The staff has reviewed these résumés and provides a summary of the experience of key plant personnel below.

Although the Plant Manager has been with LP&L for only about 1 year, he brings to the Waterford project 6 years of naval nuclear experience and almost 10 years of commercial nuclear power plant experience of which about 7 years was "hot" operating experience.

The Assistant Plant Manager-Operations and Maintenance joined the Waterford project about 1 year ago. His experience includes 7 years in the naval nuclear program and 9 years in the operation of two different commercial nuclear power plants, including the startup of one of them. He had a senior reactor operator (SRO) license at each plant. The Operations Superintendent had broad operating experience in the naval nuclear program and 6 years in reactor operator and SRO-licensed positions in the operation of a PWR similar to that at the Waterford 3 plant. The Maintenance Superintendent served 5 years in the nuclear navy, including 4 years of submarine operation, testing, overhaul, and training. The

Shift Technical Advisor (STA) Superintendent served for 7 years aboard nuclear-powered submarines and for 3 years as a licensed reactor operator and 3 years as a senior reactor operator at two commercial operating nuclear power plants.

The Assistant Plant Manager-Technical Services has been with LP&L for 20 years. For 12 years he worked in LP&L's fossil-fueled plants. For the past 8 years, he has been Assistant Station Superintendent at Waterford 3 and has received an SRO license for Waterford 3. His Technical Support Superintendent has 12 years of broad technical experience on the Waterford project, and two of the four department heads under the Technical Support Superintendent have considerable nuclear plant startup experience. Three of the four bring from 5 to 15 years of experience each in chemistry, mechanical and nuclear engineering, and reactor physics, and the fourth had 9 years of experience aboard nuclear submarines and 4 years as an NRC inspector.

By letter dated March 22, 1984, the applicant provided a tabulation of the nuclear power plant experience of licensed plant operating personnel and of the STAs.

Currently, there are five Shift Supervisors who are SRO licensed on Waterford 3. One of these has had almost 3 years of RO experience and almost 2 years of SRO experience at another commercial PWR plant and has been at the Waterford 3 plant for more than 2½ years. Three of the other four have had from 5 to 8 years of experience in the naval nuclear program, and the fourth has over 5 years of experience in naval nuclear operations. These four have been with the Waterford project for more than 6 years and have participated in operations at another commercial nuclear power plant for 2½ months. Two of the five have participated in at least one plant startup and shutdown.

Nine of ten control room supervisors have SRO licenses for Waterford. The tenth is expected to be licensed in June 1984. The nine have naval nuclear experience of about 2 to 8 years, and three of these nine have from 6 months to over 4 years of experience as ROs at another commercial PWR plant. The tenth has over 9 years of experience as an SRO/RO at another commercial PWR plant.

There are four individuals who are SRO licensed on Waterford but are assigned to RO positions on shift. All have naval nuclear experience ranging from 2 to 6 years.

There are 14 licensed reactor operators. All but one have naval nuclear experience, and all have participated in operations at another commercial nuclear power plant for 2½ months.

Four individuals have been SRO licensed on Waterford 3. All had SRO-licensed experience at other commercial PWR plants; ranging from 1 to 9 years, and three had RO-licensed experience of 1 to 2 years. These four are in staff positions but are available for serving on shift if needed.

Of the six STAs who have been hired, four have 4 years or more of experience in the naval nuclear program. All have bachelor's degrees at least in engineering or applied science

The staff has evaluated the nuclear experience of the individuals whom the applicant has identified as serving on the operating shifts. Of the 19 operations SROs, all have had "hot" plant experience of at least 6 weeks above 30% of full power, nine have experienced at least one startup and shutdown, and five have had at least 6 months of "hot" experience on shift. As long as each shift includes at least one SRO whose experience satisfies all three of the above criteria, the staff concludes that the shift staffing is acceptable. The applicant has committed to meeting this requirement.

In Supplement 2, the staff noted that the applicant planned to have a staff of 15 to 20 STAs who would serve on a 24-hour duty-day basis. FSAR Amendment 34 revises that plan: There will be seven STAs and they will be assigned to each operating shift crew. The staff finds this acceptable.

On the basis of its review of the changes to the plant organization and staffing, the staff concludes that the organization, staffing levels, and staff qualifications are adequate for operation of the facility.

13.2 Training

13.2.1 Corporate and Plant Staff Training Program

Fire Protection Training

In Supplement 5, the staff indicated that the description of the fire protection training program adequately covered the areas required by SRP Section 13.2.2 except that no commitments had been made (1) to hold meetings at least every 3 months for all brigade members to review changes in the fire protection program and (2) for all employees to participate in an annual evacuation drill. The staff further indicated that when these commitments have been made, the staff would be able to conclude that the applicant's fire protection training program was acceptable.

By letter dated February 23, 1984, the applicant submitted an extensively revised description of the fire protection training program. This revised description, which includes the applicant's commitments to the above cited requirements, fully complies with the guidelines described in SRP Section 13.2.2. Therefore, the staff considers that these open issues have been resolved.

By letter dated March 15, 1984, the applicant submitted a commitment to the requirements for fire brigade training as specified in Branch Technical Position CMEB 9.5.1, "Guidelines for Fire Protection for Nuclear Power Plants" (NUREG-0800, Section 9.5.1). This commitment was stated in the previously submitted Amendment 29 (October 20, 1982) but was inadvertently deleted from the current Amendment 34.

Training for Mitigating Core Damage

In Supplement 5, the staff indicated that the applicant had described a training program for mitigating core damage that would concentrate in the following areas:

(1) incore instrumentation
(2) incore nuclear instrumentation

(3) vital instrumentation
(4) radiation monitoring
(5) gas generation
(6) primary coolant chemistry

The subjects to be included in the training program were consistent with those required by SRP Section 13.2.2 and were, therefore, acceptable. However, the FSAR does not state clearly that STAs and operating personnel from the Plant Manager through the operations chain to the licensed operators would receive all the training indicated and that managers and technicians in the instrumentation and control, health physics, and chemistry departments would receive training commensurate with their responsibilities, as required by SRP Section 13.2.2.

Furthermore, the staff indicated that when such commitments have been made, the staff would be able to conclude that the applicant's program of training for mitigating core damage met the acceptance criteria of SRP Section 13.2.2 and was, therefore, acceptable.

By letter dated February 16, 1984, and in FSAR Amendment 34, the applicant responded to the above concern. The applicant stated that shift technical advisors and operating personnel from the Plant Manager through the operations chain to the licensed operators shall receive the training which covers all the topics specified in Enclosure 3, "Training Criteria for Mitigating Core Damage," of H. R. Denton's (NRC) letter of March 28, 1980, to all power reactor applicants and licensees. In addition, managers and technicians in the instrumentation and control, health physics, and chemistry departments shall receive training commensurate with their responsibilities. Therefore, the staff finds that the applicant's commitments are in conformance with the guidelines of SRP Section 13.2.2, satisfy the requirements of TMI Action Plan Item I.A.2.1, and are, therefore, acceptable.

Shift Technical Advisor Training

In Supplement 5, the staff indicated that the applicant had stated in the FSAR that STAs who had not functioned as STAs for 4 months or longer would be given requalification training. The 4-month period was not acceptable. The staff's acceptance criteria, as given in SRP Section 13.2.2, required such requalification training to be given to "people not actively performing the STA functions for a period of 30 days or longer." Furthermore, the staff indicated that when the applicant has provided an acceptable commitment in this area, the staff would be able to conclude that the applicant's STA training program met the acceptance criteria of SRP Section 13.2.2 and was, therefore, acceptable.

By letter dated February 16, 1984, the applicant committed that persons not actively performing STA functions for a period of 30 days or longer shall, before resuming STA activities, review the control room log to ensure they are cognizant of facility/procedural changes that had occurred during their absence. In a letter dated February 16, 1984, the applicant further committed that those persons not actively performing STA functions for a period of 30 days or longer shall, before resuming STA activities, also review the Required Reading Book to ensure that they are cognizant of facility procedural changes that occurred during their absence. The Required Reading Book is a compilation of facility and procedural changes relevant to plant operations. Furthermore, persons not performing STA functions for a period of 4 months or longer shall be recertified by test before resuming STA activities. In a letter dated April 30, 1984, the applicant stated that this recertification test will address all elements of the annual STA requalification program. The staff finds that the applicant's commitments are in conformance with the guidelines of SRP Section 13.2.2, satisfy the requirements specified in NUREG-0737, Appendix C, and are, therefore, acceptable.

Conclusions

On the basis of its evaluations presented in the SER, Supplements 2 and 5, and this supplement, the staff concludes that the applicant has described a training organization and training and retraining programs for nonlicensed personnel that meet the acceptance criteria of SRP Section 13.2.2 and are, therefore, acceptable.

13.2.2 Licensed Operator Training Program

A training program for Waterford 3 licensed operators has been implemented to develop and maintain an organization fully qualified to operate the plant and maintain plant safety. This training program will supplement the individual's background education, training, and experience, provide additional knowledge, and enhance an individual's ability to perform the assigned tasks. This training program is designed to meet the requirements of 10 CFR 19, 10 CFR 50, and 10 CFR 55; RGs 1.8 and 1.149; and requirements related to the TMI Action Plan.

13.2.2.1 Cold-License Candidate Training

In a letter dated March 15, 1984, the applicant committed to an initial cold license reactor operator and senior reactor operator candidate training program that covers the following subjects, which meet the requirements of 10 CFR 55.21:

(1) fundamentals of reactor theory
(2) general design features of the reactor core
(3) mechanical design features of the reactor primary system
(4) auxiliary systems that affect the facility
(5) general plant operating characteristics
(6) plant instrumentation and control systems
(7) plant protection systems
(8) engineered safety systems
(9) operating procedures
(10) radiation control and safety

The applicant also has committed to a training program that includes the following subjects, which meet the requirements of 10 CFR 50.22:

(1) conditions and limitations in the facility license

(2) design and operating limitations in the Technical Specifications

(3) facility license procedures for design and operating changes

(4) radiation hazards

(5) reactor theory

(6) specific operating characteristics (including coolant chemistry and causes and effects of temperature, pressure, and reactivity changes)

(7) procedures and limitations involved in initial core loading, alterations in core configurations, control rod programming, and determination of various internal and external effects on core reactivity

(8) fuel handling facilities and procedures

(9) procedures and equipment available for the handling and disposal of radioactive materials and effluents

The cold-license candidates also receive training in the following subjects, as specified in Enclosure 2 of H. R. Denton's March 28, 1980, letter:

(1) heat transfer
(2) fluid flow
(3) thermodynamics

The cold-license reactor operator and senior reactor operator candidate training program also includes the following courses:

(1) a 5-week academic refresher course

(2) a 7- to 10-week basic nuclear fundamentals course

(3) a 3-week research reactors course

(4) a 10-week observation course conducted at Florida Power and Light, St. Lucie Station, or Arkansas Power and Light, Arkansas Nuclear One, Unit 2

(5) an 8-week simulator course conducted by the Combustion Engineering PWR Nuclear Training Department.

(6) a 5-week plant-specific nuclear steam supply system lecture series course

The cold-license candidates also undergo the following training:

(1) plant systems training (13 weeks)
(2) operating characteristics training (2 weeks)
(3) procedures and Technical Specifications training (2 weeks)
(4) training related to the emergency plan (2 weeks)

13.2.2.2 On-the-Job Training

On-the-job training is provided in cold hydrostatic testing, hot functional testing, startup tests, and system walkdown checklist completion, including low-power tests for training and initial operations.

13.2.2.3 Training for Mitigating Core Damage

In a letter dated March 15, 1984, the applicant has committed to a program of training for mitigating core damage that meets the requirements specified in

Enclosure 3 of H. R. Denton's March 28, 1980, letter. The topics included in the training program are:

(1) incore instrumentation
(2) excore nuclear instrumentation (NIS)
(3) vital instrumentation
(4) primary chemistry
(5) radiation monitoring
(6) gas generation

13.2.2.4 Program Evaluation

The applicant has indicated that the effectiveness of these training programs will be evaluated through written or oral examinations, practical demonstration of skills, observation of job performance, and other methods appropriate to the subject. Results will be used in determining remedial training for the employee and as feedback to the program.

13.2.2.5 Requalification Training Program

In a letter dated March 15, 1984, the applicant has committed to a requalification training program for all licensed reactor operators and senior reactor operators. The purpose of this program is to maintain proficiency of the Waterford 3 operating organization, particularly in response to abnormal and emergency situations. Personnel holding a reactor operator or senior reactor operator license will begin this training within 3 months following receipt of their license and continue the training at 2-year intervals. This program will consist of the following:

(1) Lecture

Preplanned lecture content will be based on the results of the annual reactor operator and senior reactor operator examination that will indicate the scope and depth of coverage needed in the following subject areas:

(a) theory and principles of operation
(b) general and specific plant operating characteristics
(c) plant instrumentation and control systems
(d) plant protection systems
(e) engineered safety systems
(f) normal, abnormal, and emergency operating procedures
(g) radiation control and safety
(h) Technical Specifications
(i) applicable portions of 10 CFR, Chapter I

(2) On-the-Job-Training

The on-the-job training portion of the requalification program will consist of the following segments:

(a) Control Manipulations

The applicant has indicated in a letter dated March 27, 1984, that each licensed reactor operator is required to manipulate facility controls through at least 10 evolutions and each licensed senior operator is required to manipulate, direct or evaluate the manipulation of controls by others through the same number of plant evolutions from any combination of the following evolutions:

- manipulations to be performed annually:

  - reactor startups

  - manual control of steam generator level and/or feedwater flow rate during startup and shutdown

  - any significant (10%) power changes in manual rod control mode

  - loss of coolant including:

    1. significant steam generator leaks
    2. leaks inside primary containment
    3. large and small breaks, including leak-rate determination
    4. saturated reactor coolant response

  - loss of core coolant flow and/or natural circulation

  - loss of all feedwater flow (normal and emergency)

- manipulations to be performed on a 2-year cycle:

  - plant shutdown

  - boration and/or dilution during power operation

  - loss of instrument air supply

  - loss of electrical power and/or degraded power sources

  - loss of condenser vacuum

  - loss of service water flow if required for safety

  - loss of shutdown cooling capability

  - loss of component cooling system or cooling water to an individual component

  - loss of normal feedwater flow or normal feedwater system failure

  - loss of protective system channel

  - mispositioned control rod or rods (or rod drops)

  - inability to drive control rods

  - conditions requiring use of emergency boration

- fuel cladding failure or high activity in reactor coolant

- turbine or generator trip

- malfunction of automatic control system(s) that affect reactivity

- malfunction of reactor coolant pressure and/or volume control system

- reactor trip

- main steamline break (inside or outside containment)

- nuclear instrumentation failure(s)

The above manipulations are acceptable in meeting the requirements specified in Enclosure 4 of H. R. Denton's letter of March 28, 1980.

(b) Knowledge of Facility Design, Procedure, and License Change and Abnormal and Emergency Procedures

To ensure a continuing awareness of the actions and responses necessary during abnormal and emergency situations, each licensed reactor operator and senior reactor operator will be kept aware of the following items:

· facility design changes
· facility procedure changes
· Technical Specification changes
· emergency preparedness plan
· radiation control procedures
· operating procedures
· abnormal and emergency procedures
· significant operational events

(3) Simulator Training

The applicant has indicated that licensed reactor operators and senior reactor operators will participate in simulator training during their requalification program. Simulator training will be provided by the Combustion Engineering PWR Nuclear Training Department or through an equivalent arrangement. This program provides hands-on experience for personnel and complies with the intent of RG 1.149. In 1987, Louisiana Power & Light Company will install a Waterford 3 plant-specific nuclear power simulator. This simulator will meet the intent of the applicable design requirements of RG 1.149. The applicant has committed to the requirement specified in Enclosure 1 of H. R. Denton's letter of March 28, 1980, which requires all licensed operators to participate in a simulator training program as part of the requalification program.

(4) Evaluation

As described in Appendix A to 10 CFR 55, the evaluation program for licensed personnel shall include the following:

(a) Annual Written Examination

The applicant has indicated that annual examinations will be given to determine areas in which retraining is needed to upgrade licensed reactor operator and senior reactor operator knowledge.

(b) Systematic Observation and Evaluation

The applicant has indicated that there will be systematic observation and evaluation of the performance and competency of licensed reactor operators and senior reactor operators. This will include observation and evaluation of actions taken or to be taken during actual or simulated abnormal and emergency conditions.

(5) Accelerated Requalification Program

The applicant has indicated in a letter dated March 27, 1984, that any licensed reactor operator or senior reactor operator shall be entered into an accelerated requalification program because of failure of any requirement of the requalification program.

(6) Records

The applicant has indicated that records of the requalification training program shall be maintained for a period of 2 years from the date of the recorded event. These records shall contain copies of examinations, answers, results of evaluations, and documentation of any additional training administered in areas in which a licensed reactor operator or senior reactor operator has exhibited deficiencies.

13.2.2.6 TMI-Related Requirements for New Operating License

I.A.2.1 Immediate Upgrading of Reactor Operator and Senior Reactor Operator Training and Qualification

The applicant has established a program to ensure that all reactor operator and senior reactor operator (SRO) license candidates have the prescribed experience, qualification, and training. SRO license candidates who possess a degree in engineering or applicable science are considered to meet the 1-year experience requirement as a reactor operator provided they (1) satisfy the requirements in Sections A.1.a and A.2 of Enclosure 1 to the letter from H. R. Denton to all power reactor applicants and licensees dated March 28, 1980, and (2) have participated in a training program equivalent to that of a cold-license SRO candidate.

As an applicant for an operating license, Waterford 3 is not subject to the 1-year experience requirement for cold-license SRO candidates. However, after 1 year of station operation, individuals applying for an SRO license will be required to comply with the 1-year experience requirement for hot-license SRO candidates, unless they have previous experience in an equivalent position at another nuclear plant or at a military propulsion reactor. The experience of license candidates in the latter category will be documented by the applicant on a case-by-case basis in sufficient detail so that the staff can make a finding regarding equivalency.

Also, the requirement for 3-months of onshift experience for control room operators and SRO candidates as an extra person on shift is not required for cold-license candidates and, hence, is not applicable to Waterford 3. However, the applicant will be required to comply with this requirement for hot-license candidates after 3 months of plant operation.

The applicant's training program includes the following topics: heat transfer, fluid flow, and thermodynamics. This is in accordance with Enclosure 2 of H. R. Denton's letter of March 28, 1980.

The applicant's training program includes topics pertaining to training for mitigating core damage. This is in accordance with Enclosure 3 of H. R. Denton's letter of March 28, 1980.

On the basis of its review, the staff finds that the applicant has satisfied the requirements of Item I.A.2.1 of the TMI Action Plan.

I.A.2.3 Administration of Training Programs

As specified in Enclosure 1 of H. R. Denton's March 28, 1980, letter, the staff requires that all instructors who teach systems integrated responses and transient and simulator courses shall be SRO certified and shall continue to participate in appropriate requalification programs. The applicant has made the commitment that all instructors of licensed operators for safety-related subjects and courses either will be qualified as a senior reactor operator or will complete a licensed instructor certification program. These instructors will participate in the appropriate requalification program.

On the basis of its review, the staff finds that the applicant has satisfied the requirements of Item I.A.2.3 of the TMI Action Plan.

Conclusion

On the basis of the results of this review, the staff concludes that the training program and requalification training program for licensed reactor operators and senior reactor operators meet the requirements specified in NUREG-0800, H. R. Denton's March 28, 1980, letter, and 10 CFR 55, Appendix A, and are, therefore, acceptable.

13.3 Emergency Preparedness Evaluation

13.3.1 Introduction

Subsequent to the issuance of Supplement 5, Revisions 5 and 6 to the Waterford 3 Emergency Plan were submitted by the applicant in January 1984. Revision 6 was the subject of review during a followup emergency preparedness implementation appraisal (EPIA) at Waterford 3 on January 30-February 10, 1984. Various changes occurred to Section 5 ("Emergency Organization") and Section 6 ("Emergency Response Measures") of the Plan as a result of Revisions 5 and 6. During the followup EPIA, the staff raised questions over changes to certain portions of the Plan and attempted to reach a resolution on each item with the applicant. A detailed discussion of the staff's review and evaluation of Revisions 5 and 6 to the Plan is contained in Attachment 1 to the followup EPIA report, Office of Inspection and Enforcement (IE) Report No. 50-382/84-02.

On February 21, 1984, and March 8, 1984, the applicant responded to the staff's comments on Revisions 5 and 6 as well as the unresolved Plan items that were previously identified during the initial EPIA (IE Report No. 50-382/83-08). An acceptable response has been obtained for the Plan items identified by the staff. In correspondence dated March 8, 1984, the applicant committed to make all revisions to the Plan and procedures by May 4, 1984. Revision 7 to the Plan was submitted by letter dated May 7, 1984, and the staff will provide its evaluation in a subsequent supplement to the SER.

Section 13.3.2 to this supplement provides the staff's discussion of certain items previously identified in Supplement 5. The listing of items in Section 13.3.2 of this supplement corresponds to the listing of items as they appear in Supplement 5. Section 13.3.3 contains a status report on the findings and determinations of the Federal Emergency Management Agency (FEMA) on the adequacy of offsite plans and preparedness. Section 13.3.4 provides the staff's conclusions.

13.3.2 Evaluation of the Emergency Plan

13.3.2.2 Emergency Preparedness Items Under Review

13.3.2.2.1 Emergency Classification System

The applicant has revised Section 4.0 to the Plan, and the emergency action level scheme now conforms to the guidance criteria of Appendix 1 to NUREG-0654. On the basis of its review of the Plan, the staff finds that the applicant has provided an acceptable response to this item.

13.3.2.4 Alert and Notification System

Following a sound level test of selected sirens on November 4-5, 1983, the applicant determined that 10 additional sirens were required to provide adequate coverage of the 10-mile emergency planning zone with 5 of these allocated for areas of future population expansion. The applicant currently is procuring these sirens.

Revision 5 to the Plan provided new information on the siren alerting system, including alerting augmentation by tone-alert receivers, helicopters, fan-out warning teams, the St. Charles Industrial Hot Line, and the Industrial Mutual Aid Radio System for St. John the Baptist Parish. During the followup EPIA, the staff requested additional information on the tone-alert receivers. In correspondence dated January 21, 1984, the applicant committed to provide this information in a Plan revision by May 4, 1984, as a part of a complete description of the primary and backup alerting system and the public notification system. Revision 7 to the Plan was submitted by letter dated May 7, 1984, and the staff will provide its evaluation in a subsequent supplement to the SER.

The staff will require the applicant to demonstrate that the systems are installed and operational before fuel loading.

13.3.3 Status Report on the FEMA Evaluation of Offsite Emergency Plans

The State of Louisiana Peacetime Radiological Response Plan, Revision 4, and the St. Charles and St. John the Baptist Parishes' emergency response plans have been reviewed by FEMA. Interim findings by FEMA are provided in Appendix E to this supplement. FEMA concludes that there is reasonable assurance that the plans are adequate and capable of being implemented in the event of an accident at the Waterford site. Included in its interim findings are FEMA's response with regard to the concerns of the St. John the Baptist Parish Civil Defense Director and comments on the E. L. Quarantelli report entitled "Evacuation Behavior: Case Study of the Taft Louisiana Chemical Tank Explosion Incident," May 1983.

Supplemental FEMA findings are required regarding the resolution of certain offsite planning issues specified in the Atomic Safety Licensing Board Partial Initial Decision of November 3, 1982, as amended by Memorandum and Order dated December 14, 1982.

A full-scale emergency preparedness exercise involving participation by the utility, State, and local emergency response organizations was held on February 8, 1984. Following the receipt of FEMA's supplemental findings and its report on the offsite preparedness exercise, the staff will provide the results of its review of FEMA's findings and determinations on the state of offsite emergency preparedness.

13.3.4 Conclusions

In Supplement 5, the staff concluded that, subject to confirmation of certain items committed to by the applicant, the state of onsite emergency preparedness provides reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency. On the basis of a review of Revisions 5 and 6 to the Waterford 3 Emergency Plan and the applicant's commitments expressed in correspondence dated February 21, 1984, and March 8, 1984, the staff concludes that, subject to satisfactory completion of Plan changes committed to by the applicant, the Plan will continue to meet the requirements of 10 CFR 50 and the guidance criteria in NUREG-0654, Revision 1. The staff's review of the supplemental findings by FEMA regarding the state of offsite plans and preparedness will be provided in a subsequent supplement.

13.4 Review and Audit

In Supplement 2, the staff concluded that the applicant had taken adequate steps to provide for safety review and audit of plant operations, including the establishment of an Independent Safety Engineering Group that met the requirements of TMI Action Plan Item I.B.1.2 of NUREG-0737.

In FSAR Amendment 30, the applicant proposed a significant change to the functions of the corporate Safety Review Committee and the Independent Safety Engineering Group. The staff and the applicant have discussed these changes during the process of preparing Section 6, "Administrative Controls," of the Technical Specifications. The applicant has indicated that the Standard Technical Specifications that address these matters are acceptable. Therefore, this is no longer an open item.

13.6 Physical Security

13.6.1 Introduction

The applicant has filed with the NRC the following security program plans: "Site Security Plan Waterford Steam Electric Station Unit No. 3," "Waterford 3 Steam Electric Station Safeguards Contingency Plan," and "Waterford Generating Station Guard Training and Qualification Plan."

On the basis of a review of the subject documents and visits to the site, the staff has concluded that the protection provided by the applicant against radiological sabotage at Waterford 3 meets the requirements of 10 CFR 73. Accordingly, the protection will ensure that the health and safety of the public will not be endangered.

13.6.2 Physical Security Organization

To satisfy the requirements of 10 CFR 73.55(b), the applicant has provided a physical security organization that includes a Shift Supervisor who is on site at all times with the authority to direct the physical protection activities. To implement the commitments made in the physical security plan, training and qualification plan, and the safeguards contingency plan, written security procedures specifying the duties of the security organization members have been developed and are available for inspection. The training program and critical security tasks and duties for the security organization personnel are defined in the "Waterford Generating Station Guard Training and Qualification Plan," which meets the requirements of 10 CFR 73, Appendix B, for the training, equipping, and requalification of the security organization members. The physical security plan and the training program provide commitments that preclude the assignment of any individual to a security-related duty or task before the individual is trained, equipped, and qualified to perform the assigned duty in accordance with the approved guard training and qualification plan.

13.6.3 Physical Barriers

In meeting the requirements of 10 CFR 73.55(c), the applicant has provided a protected area barrier that meets the definition in 10 CFR 73.2(f)(1). An isolation zone, to permit observation of activities along the barrier, of at least 20 ft is provided on both sides of the barrier with the exception of the locations listed in the protected appendix that is part of this evaluation. The staff has reviewed those locations and determined that the security measures in place are satisfactory and continue to meet the requirements of 10 CFR 73.55(c).

Illumination of 0.2 ft-candle is maintained for the isolation zones, protected area barrier, and external portions of the protected area. In areas where illumination of 0.2 ft-candle cannot be maintained, special procedures are applied as described in the protected appendix.

13.6.4 Identification of Vital Areas

The design bases for the applicant's program for identifying vital equipment included the regulatory definition of vital, 10 CFR 100 dose guidelines, and the criteria contained in RG 1.29 and Review Guideline #17 (transmitted by memorandum dated January 23, 1978). The program used conservative assumptions (i.e., no credit is given for offsite power and equipment not protected as vital is assumed to be unavailable) and a detailed plant analysis (fault tree) to identify both single items of equipment and combinations thereof that require protection. The protected appendix contains a discussion of the applicant's program and identifies those areas and equipments determined to be vital.

Vital equipment is located within vital areas which are located within the protected area and which require passage through at least two barriers, as defined in 10 CFR 73.2(f)(1) and (2), to gain access to the vital equipment. Vital area barriers are separated from the protected area barrier. In addition to the dual-barrier system required by 10 CFR 73, the applicant has provided, for other reasons, a third barrier between the protected area barrier and the vital area barrier.

Patrols of the protected area are performed at random intervals to detect the presence of unauthorized persons, vehicles, and materials.

The control room and central alarm station are provided with bullet-resistant walls, doors, ceilings, floors, and windows.

On the basis of these findings and the analysis in Paragraph D of the protected appendix, the staff has concluded that the applicant's program for identification of vital equipment satisfies the regulatory intent. However, this program is subject to onsite validation by the staff in the future and to subsequent changes if they are found to be necessary.

13.6.5 Access Requirements

In accordance with 10 CFR 73.55(d), all points of personnel and vehicle access to the protected area are controlled. The individual responsible for controlling the final point of access into the protected area is located in a bullet-resistant structure. As part of the access control program, vehicles (except under emergency conditions), personnel, packages, and materials entering the protected area are searched for explosives, firearms, and incendiary devices by electronic search equipment and/or physical search.

Vehicles admitted to the protected area, except applicant-designated vehicles, are controlled by escorts. Applicant-designated vehicles are limited to onsite station functions and remain in the protected area except for operational maintenance, repair, security, and emergency purposes. Positive control over the vehicles is maintained by personnel authorized to use the vehicles or by the escort personnel.

A picture badge/key card system, using encoded information, identifies individuals who are authorized unescorted access to protected and vital areas and is used to control access to these areas. Individuals not authorized unescorted access are issued non-picture badges, which indicate an escort is required. Access authorizations are limited to those individuals who need access to perform their duties.

Unoccupied vital areas are locked and alarmed. During periods of refueling or major maintenance, access to the reactor containment(s) is positively controlled by a member of the security organization to ensure that only authorized individuals and materials are permitted to enter. In addition, all doors and personnel/equipment hatches into the reactor containment(s) are locked and alarmed. Keys, locks, combinations, and related equipment are changed on an annual basis. In addition, when an individual's access authorization has been terminated because of a lack of reliability or trustworthiness or because of poor work performance, the keys, locks, combinations, and related equipment to which that person had access are changed.

13.6.6 Detection Aids

In satisfying the requirements of 10 CFR 73.55(e), the applicant has installed intrusion detection systems at the protected area barrier, at entrances to vital areas, and at all emergency exits. Alarms from the intrusion detection system annunciate within the continuously manned central alarm station and a secondary alarm station located within the protected area. The central alarm station is located so that the interior of the station is not visible from outside the perimeter of the protected area. In addition, the central station is constructed so that walls, floors, ceilings, doors, and windows are bullet resistant. The alarm stations are located and designed so that a single act cannot interdict the capability of calling for assistance or responding to alarms. No functions or duties that would interfere with its alarm response function are performed in the central alarm station. The intrusion detection system transmission lines and associated alarm annunication hardware are self checking and can indicate if they have been tampered with. Alarm annunicators indicate the type of alarm and its location when activated. An automatic indication of when the alarm system is on standby power is provided in the central alarm station.

13.6.7 Communications

As required in 10 CFR 73.55(f), the applicant has provided for the capability of continuous communications between the central and secondary alarm station operators, guards, watchmen, and armed response personnel through the use of a conventional telephone system and a security radio system. In addition, direct communication with the local law enforcement authorities is maintained through the use of a conventional telephone system and two-way FM radio links. All nonportable communication links, except the conventional telephone system, are provided with an uninterruptible emergency power source.

13.6.8 Test and Maintenance Requirements

In meeting the requirements of 10 CFR 73.55(g), the applicant has established a program for the testing and maintenance of all intrusion alarms, emergency alarms, communication equipment, physical barriers, and other security-related devices and equipment. Equipment or devices that do not meet the design performance criteria or have failed to otherwise operate will be compensated for by appropriate measures as defined in the "Waterford Steam Electric Station Unit 3 Physical Security Plan" and in site procedures. The compensatory measures defined in the plan and the procedures will ensure that the effectiveness of the security system is not reduced by failures or other contingencies affecting the operation of the security-related equipment or structures. Intrusion detection systems are tested for proper performance at the beginning and end of any period during which they are used for security purposes. Such testing will be conducted at least once every 7 days.

Communication systems for onsite communications are tested at the beginning of each security shift. Offsite communication systems are tested at least once each day.

Audits of the security program are conducted once every 12 months by personnel independently of site security management and supervision. The audits, focusing on the effectiveness of the physical protection provided by the onsite security organization implementing the approved security program plans, include, but are not limited to, a review of the security procedures and practices, system testing and maintenance programs, and local law enforcement assistance agreements. LP&L quality assurance and corporate security personnel will prepare a report documenting audit findings and recommendations.

13.6.9 Response Requirements

In meeting the requirements of 10 CFR 73.55(h), the applicant has provided for armed responders who will be immediately available for response duties on all shifts, consistent with the requirements of the regulations. Considerations used in support of the required number are given in the protected appendix. In addition, liaison with local law enforcement authorities to provide additional response support in the event of security events has been established and documented.

The applicant's safeguards contingency plan for dealing with thefts, threats, and radiological sabotage satisfies the requirements of 10 CFR 73, Appendix C. The plan identifies security events that could initiate radiological sabotage and identifies the appliant's preplanning, response resources, safeguards contingency participants, and coordination activities for each identified event. Through this plan, upon the detection of abnormal presence or activities within the protected or vital areas, response activites using the available resources would be initiated. The response activities and objectives include the neutralization of the existing threat by requiring the response force members to interpose themselves between the adversary or adversaries and their objective, instructions to use force commensurate with that used by the adversary or adversaries, and authority to request sufficient assistance from the local law enforcement authorities to maintain control over the situation.

To assist in the assessment/response activities, a closed-circuit television system, with the capability to observe the entire protected area perimeter, isolation zones, and a majority of the protected area, is provided to the security organization.

13.6.10 Employee Screening Program

In meeting the requirements of 10 CFR 73.55(a) to protect against the design-basis threat as stated in 10 CFR 73.1(a)(1)(ii), the applicant has provided an employee screening program. Personnel who successfully complete the employee screening program or its equivalent may be granted unescorted access to protected and vital areas at the Waterford site. All other personnel requiring access to the site are escorted by persons authorized and trained for escort duties and who have successfully completed the employee screening program.

The employee screening program is based on accepted industry standards and includes a background investigation, a psychological evaluation, and a continuing observation program. In addition, the applicant may recognize the screening program of other nuclear utilities or contractors on the basis of a comparability review conducted by LP&L. The program also provides for a "grandfather clause" exclusion, which allows recognition of a certain period of trustworthy service with the utility or contractor as being equivalent to the overall employee screening program.

The staff has reviewed the applicant's employee screening program against the accepted industry standards (American National Standards Institute (ANSI) N18.17 1973) and has determined that the program is acceptable.

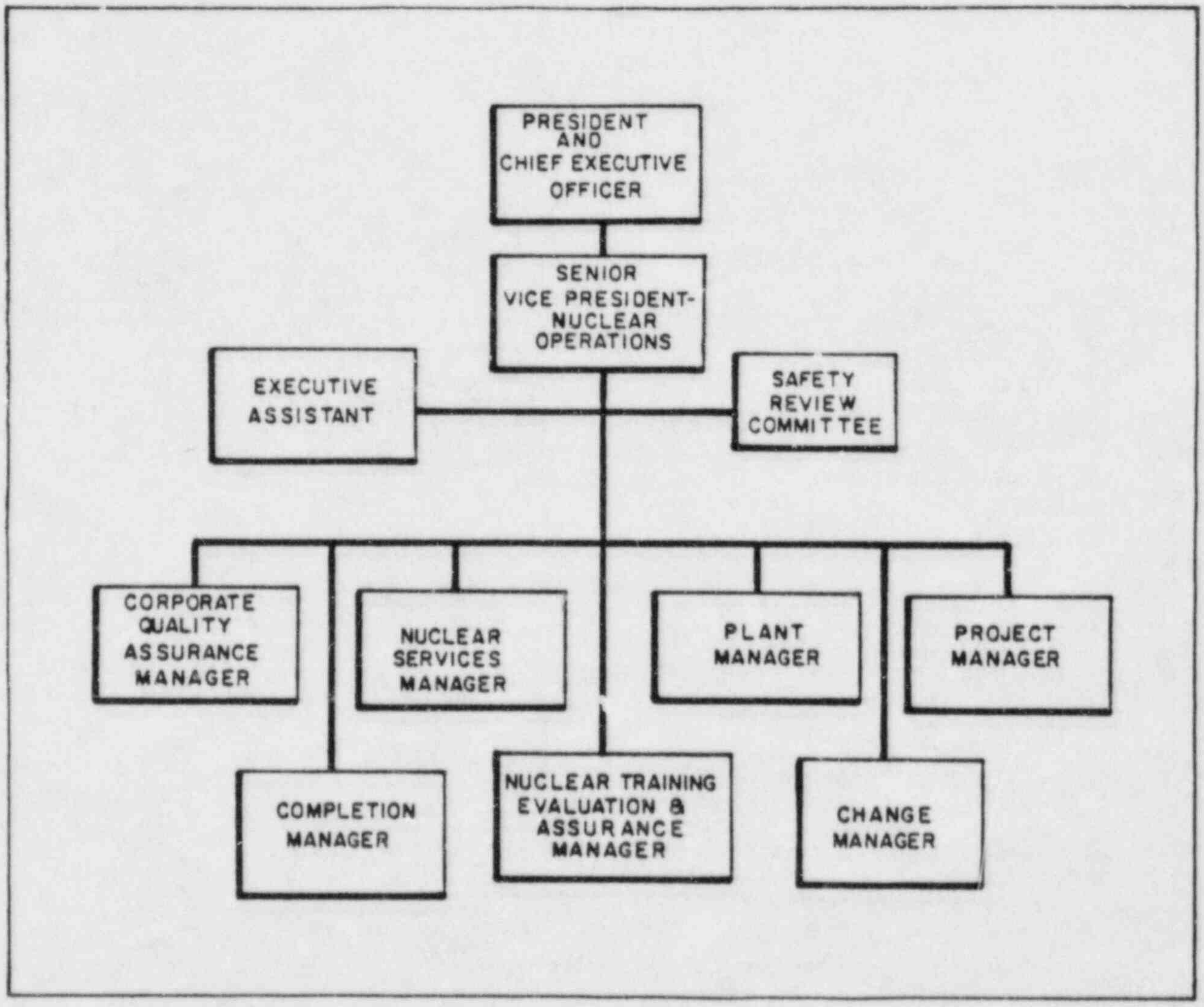


Figure 13.1 Nuclear Operations Organization
Source: FSAR Figure 13.1-2 (Amendment 34)

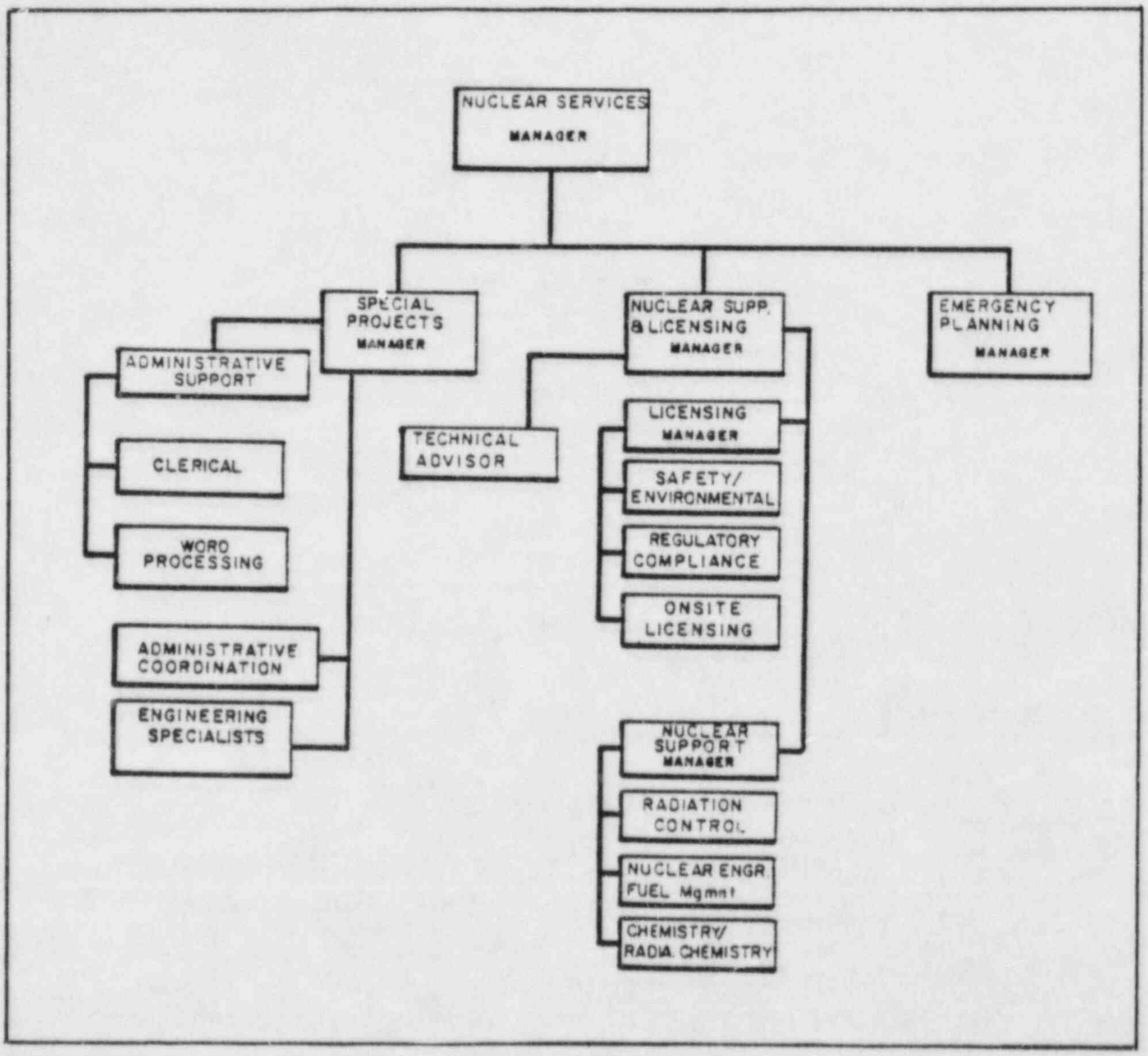


Figure 13.2 Nuclear Services Organization
Source: FSAR Figure 13.1-3 (Amendment 34)

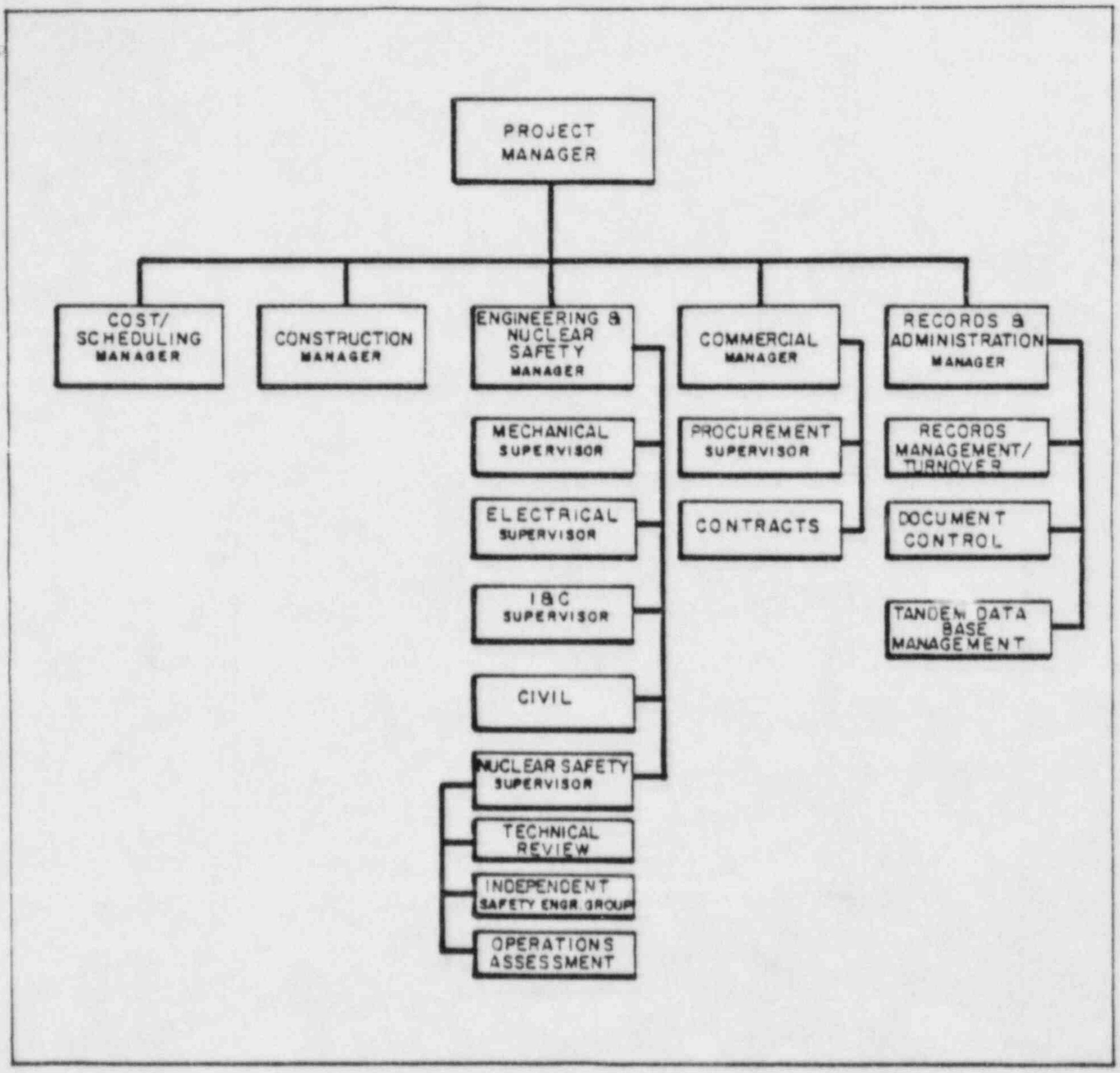


Figure 13.3 Project Management
Source: FSAR Figure 13.1-4 (Amendment 4)

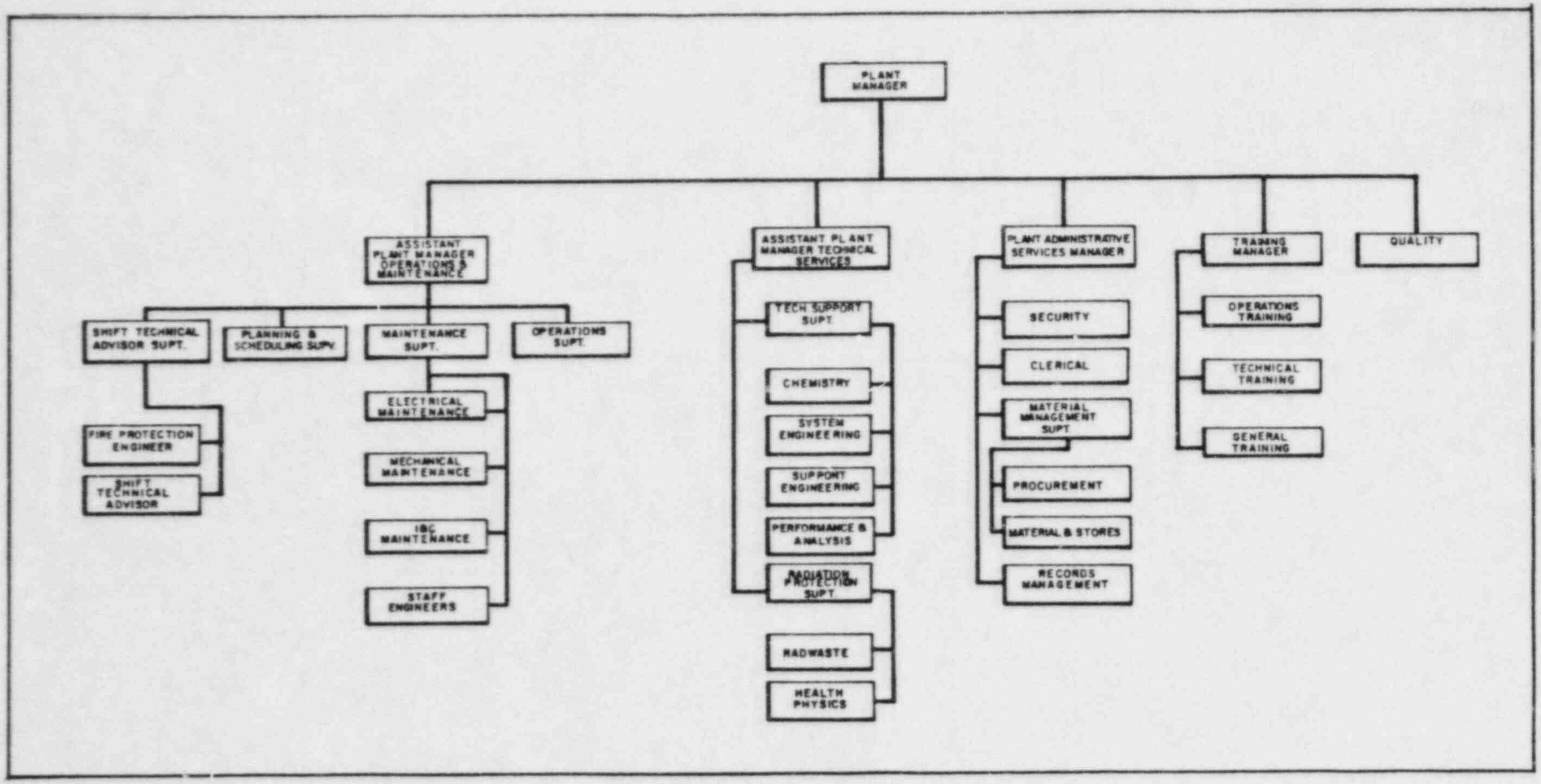
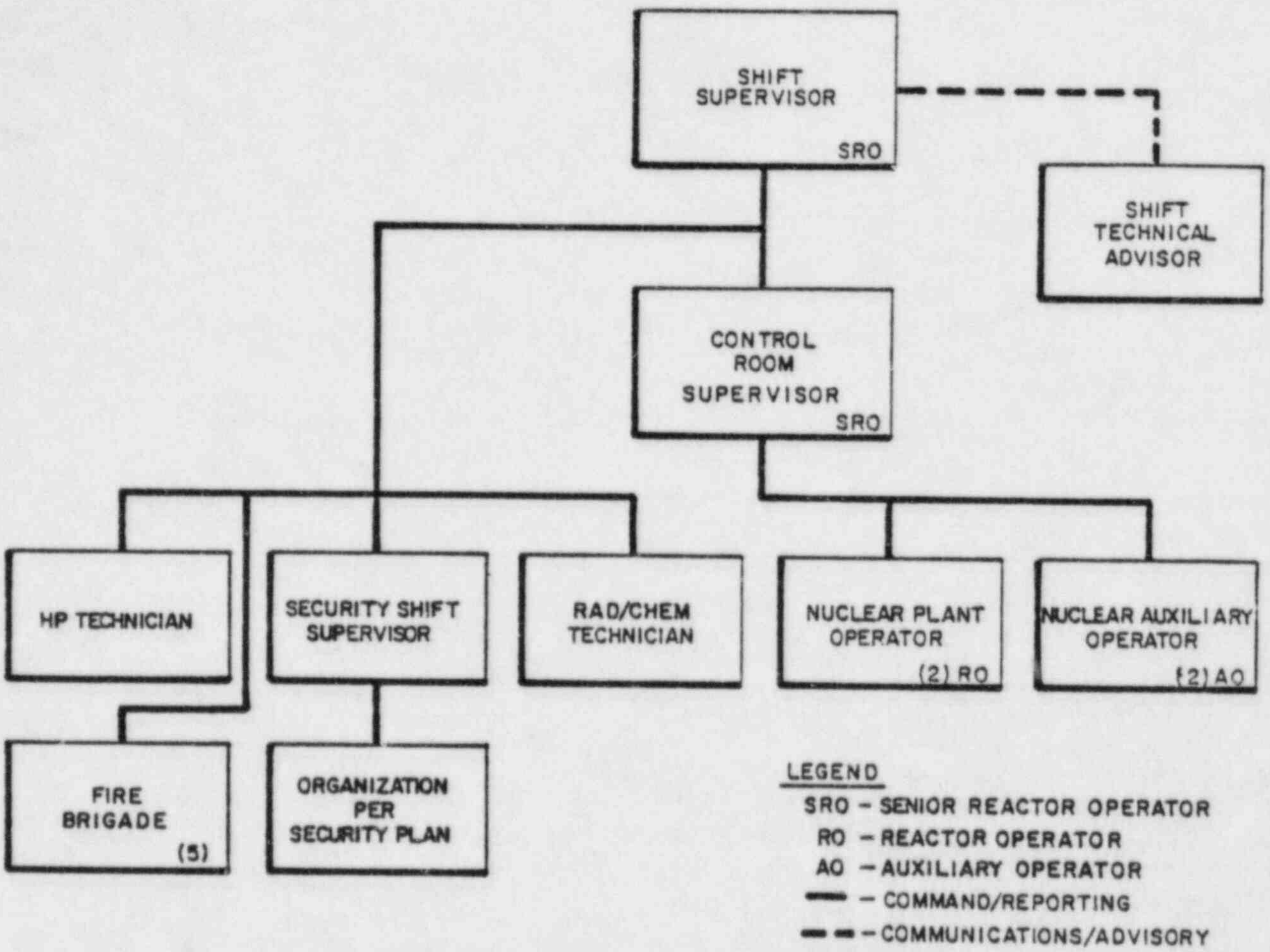


Figure 13.4 Plant Organization
Source: FSAR Figure 13.1-5 (Amendment 34)



* This figure depicts the normal expected shift organization. The minimum shift shall be at least that specified by the technical specifications.

Figure 13.5 Typical Waterford 3 Shift Organization - Modes 1, 2, 3, and 4

Table 13.1 Nuclear Operations Staffing

Staff	Approximate authorized staffing level
Nuclear Operations	
Staff	4
Plant Operations	
Staff	2
Plant Quality	13
Training	31
Plant Technical Services	3
Administrative Services	69
Technical Support	45
Radiation Protection	49
Operations	73
Staff Technical Advisor	10
Maintenance	120
Planning and Scheduling	7
Total	422
Nuclear Services	
Staff	2
Special Projects	18
Nuclear Support and Licensing	25
Emergency Planning	5
Total	50
Project Management	
Construction	8
Engineering and Nuclear Safety	40
Commercial	10
Cost and Schedule	7
Records and Administration	29
Total	94
Training Evaluation and Assurance	
Staff	5
Completion Management	
Staff	15
Change Management	
Staff	1
Quality Assurance	
Staff	46
Total	633

14 INITIAL TEST PROGRAM

In a letter dated March 10, 1983, the applicant submitted a proposal to perform special natural circulation testing and training in conjunction with the loss-of-flow test. FSAR Section 14.2.12.3.15 specifies that natural circulation testing and training be performed during low-power operation (mode 2) in accordance with the staff position forwarded to the applicant in a letter dated November 14, 1980. The applicant's letter of March 10, 1983, proposes that the natural circulation testing and training be conducted in mode 3 under conditions of actual decay heat removal following a reactor trip from 80% of rated power level. This planned trip occurs during the loss-of-flow test described in FSAR Section 14.2.12.3.34.

The staff has evaluated the applicant's proposal and has concluded that the testing and training objectives stated in the staff's position can be readily accomplished during mode 3 natural circulation operation if sufficient decay heat is available. The adequacy of the post 80% power trip condition for demonstrating natural circulation was previously determined acceptable for San Orofre Unit 2 (NUREG-0712, Supplement 1).

Because of the similarity between Waterford and San Onofre Unit 2, sufficient decay heat should be available to meet the testing and operator training objectives of the test. Additional operating experience with natural circulation also will be acquired during the loss-of-offsite-power test (FSAR Section 14.2.12.3.35) and the simulated loss-of-all-AC test (FSAR Section 14.2.12.3.41). On this basis the applicant's March 10, 1983, proposal is acceptable.

Chapter 14 of the Waterford FSAR has been revised to include the proposal contained in the March 10, 1983, letter and evaluated above. The staff's review of the applicant's revised FSAR Chapter 14 will be provided in a subsequent supplement to the SER.

17 QUALITY ASSURANCE

17.1 General

The description of the quality assurance (QA) program for the operations phase of Waterford 3 is contained in FSAR Section 17.2. Staff evaluation of this QA program through FSAR Amendment 34 is based on a review of this information and discussions with representatives of LP&L. NRC assessed LP&L's QA program for the operations phase to determine if it complies with the requirements of 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," the applicable QA-related regulatory guides listed in Table 17.1, and SRP Section 17.2, "Quality Assurance During the Operations Phase" (NUREG-0800).

17.2 Organization

The structure of the organization responsible for the operation of Waterford 3 and for the establishment and execution of the operations phase QA program is shown in Figure 17.1.

The Senior Vice President-Nuclear Operations has overall responsibility for the Waterford 3 plant, including defining QA policies. Reporting to the Senior Vice President-Nuclear Operations are the Nuclear Services Manager, Project Manager-Nuclear, Completion Manager-Nuclear, Plant Manager-Nuclear, Safety Review Committee, and the Corporate Quality Assurance Manager.

The Corporate QA Manager, under the direction of the Senior Vice President-Nuclear Operations, is responsible for developing, coordinating, and ensuring implementation of the LP&L QA program. The offsite Engineering and Systems Development QA Manager and the onsite Nuclear Operations QA Manager report to the Corporate QA Manager.

The offsite Engineering and Systems Development QA Manager's responsibilities include developing and maintaining QA policies and procedures; conducting surveillance and audits of contractors and vendors to verify compliance with applicable requirements; reviewing drawings, procedures, and specifications to ensure proper inclusion of QA requirements; and analyzing conditions adverse to quality for quality trends.

The onsite Nuclear Operations QA Manager's responsibilities include reviewing procurement documents to ensure proper inclusion of QA requirements; reviewing design drawings and specifications, and changes thereto, to ensure that the documents are prepared, reviewed, and approved in accordance with applicable procedures and that they contain the necessary QA requirements; and monitoring plant activities to verify compliance with applicable requirements.

The Plant Manager-Nuclear is responsible for the operation and maintenance of Waterford 3, which includes the onsite implementation of the QA program. The Plant Quality Manager reports directly to the Plant Manager-Nuclear and is responsible for reviews of instuctions, procedures, drawings, and procurements to ensure proper inclusion of QA requirements; receipt inspections; nondestructive examinations; inspections and/or verifications; identification, segregation, review, and disposition of nonconforming materials, parts, components, and services; and preparation of inspection instructions.

The onsite Plant Quality Manager, the Corporate Quality Assurance Manager, and their staffs have the authority and organizational freedom to identify quality problems; initiate, recommend, or provide problem solutions through designated channels and verify implementation of satisfactory solutions; and stop or control further processing, delivery, or installation of nonconforming material.

17.3 Quality Assurance Program

The QA program for the operation of Waterford 3 describes the QA policies, goals, objectives, and requirements to be implemented at the station to ensure that safety-related activities are performed in a controlled manner and documented to provide objective evidence of compliance with NRC regulations and guidance. The QA program is implemented by the Quality Assurance Manual, which includes the QA policies, procedures, and instructions. These documents present the detailed techniques and methods by which the requirements of Appendix B to 10 CFR 50 and the provisions of the NRC regulatory guidance shown in Table 17.1 are satisfied. They are reviewed and concurred in by the Corporate QA Manager.

The QA program requires that QA documents encompass detailed controls for (1) translating codes, standards, and regulatory requirements into specifications, procedures, and instructions; (2) developing, reviewing, and approving procurement documents, including changes; (3) prescribing all quality-affecting activities by documented instructions, procedures, or drawings; (4) issuing and distributing approved documents; (5) purchasing items and services; (6) identifying materials, parts, and components; (7) performing special processes; (8) inspecting and/or testing material, equipment, processes, or services; (9) calibrating and maintaining measuring and test equipment; (10) handling, storing, and shipping items; (11) identifying the inspection, test, and operating status of items; (12) identifying and dispositioning nonconforming items; (13) correcting conditions adverse to quality; (14) preparing and maintaining QA records; and (15) auditing activities that affect quality.

The QA program requires the establishment and continuous implementation of the QA indoctrination, training, and retraining program to ensure that persons involved in safety-related activities are knowledgeable about QA instructions and implementing procedures and demonstrate a high level of competence and skill in the performance of their quality-related activities.

Quality is verified through surveillance, inspection, testing, checking, and audit of work activities using procedures, instructions, and/or checklists. Inspections are performed by inspectors who are not directly responsible for performing the actual work activity and who have been qualified and certified in accordance with codes, standards, or company training programs.

The Corporate QA Manager is responsible for the establishment and implementation of the audit program. Audits are performed with written procedures or checklists by qualified personnel not having direct responsibility in the areas being audited. The QA program establishes a comprehensive audit system to ensure that the QA program requirements and related supporting procedures are effective and properly implemented during operations. Audits include an objective evaluation of QA practices, procedures, and instructions; work areas, activities, processes, and items, the effectivenss of implementation of the QA program; and conformance with policy directives.

The QA program requires documentation of audit results and review by management having responsibility in the area audited to determine and take corrective action as required. Reaudits are performed to determine that nonconformances are effectively corrected and that the corrective action precludes repetitive occurrences. Audit findings, which indicate quality trends and the effectiveness of the QA program, are reviewed by the Corporate QA Manager and are reported to the Senior Vice President-Nuclear Operations on a regular basis.

17.4 Conclusions

On the basis of the review and evaluation of the QA program description contained in FSAR Section 17.2 for Waterford 3, the staff concludes that:

(1) The QA organization of LP&L provides sufficient independence from cost and schedule (when opposed to safety considerations), authority to effectively carry out the operations QA program, and access to management at a level necessary to perform the QA functions.

(2) The QA program describes requirements, procedures, and controls that, when properly implemented, comply with the requirements of Appendix B to 10 CFR 50 and with the acceptance criteria contained in SRP Section 17.2 (NUREG-0800).

Accordingly, the staff concludes that the applicant's description of the QA program is in compliance with applicable NRC regulations.

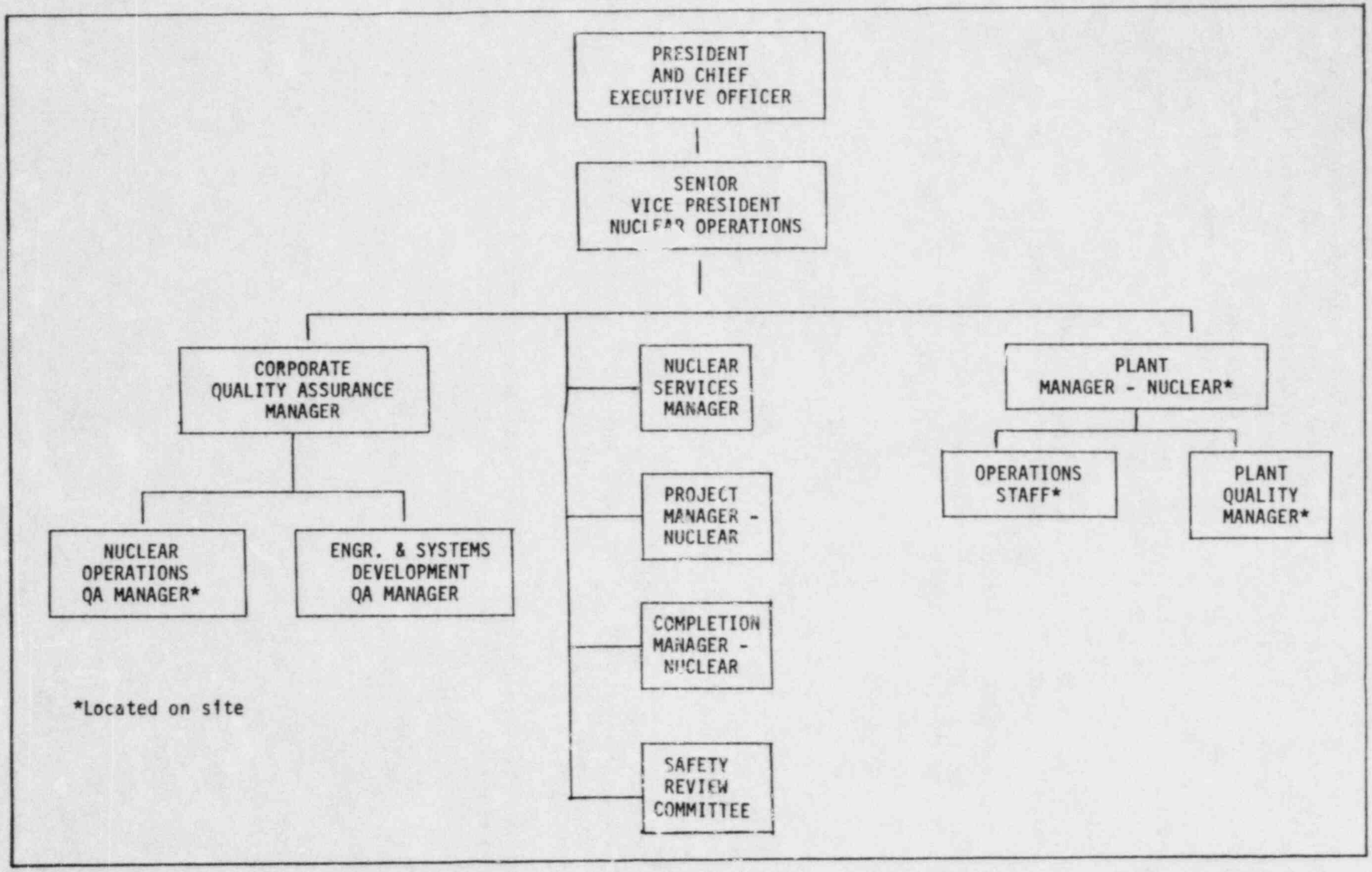


Figure 17.1 Waterford 3 organization

Table 17.1 Regulatory guides applicable to quality assurance program

Regulatory guide	Rev. no.	Date	Title
1.30	0	August 1972	Quality Assurance Requirements for Installation, Inspection, and Testing of Instrumentation and Electric Equipment
1.33	2	February 1978	Quality Assurance Program Requirements (Operation)
1.37	0	March 1973	Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants
1.38	2	May 1977	Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants
1.39	2	September 1977	Housekeeping Requirements for Water-Cooled Nuclear Power Plants
1.58	1	September 1980	Qualifications of Nuclear Power Plant Inspection, Examination, and Test Personnel
1.64	2	June 1976	Quality Assurance Requirements for the Design of Nuclear Power Plants
1.74	0	February 1974	Quality Assurance Terms and Definitions
1.88	2	October 1976	Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records
1.94	1	April 1976	Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants
1.116	0-R	June 1977	Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems
1.123	1	July 1977	Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants
1.144	1	September 1980	Auditing of Quality Assurance Programs for Nuclear Power Plants
1.146	0	August 1980	Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants

22 TMI-2 REQUIREMENTS

II.B.3 Postaccident Sampling Capability

In its safety evaluation, the staff found that the applicant's postaccident sampling system (PASS) met 8 of the 11 license conditions for Item II.B.3 in NUREG-0737. The remaining three license conditions, which were not resolved, are:

Criterion (6) Provide a procedure for relating radionuclide gaseous and ionic species to estimate core damage.

Criterion (9) Provide information on (a) testing frequency and type of testing to ensure long-term operability of the postaccident sampling system and (b) operator training requirements for postaccident sampling.

Criterion (11) Provide information on accuracy and sensitivity for analytical procedures and on-line instrumentation when exposed to an accident environment.

By letters dated September 21, 1982, July 21, 1983, August 3, 1983, November 29, 1983, and December 2, 1983, the applicant provided additional information.

The applicant provided a procedure for the prediction of core damage based on the generic CE Owners Group procedure. This procedure is to be used for analysis of radiochemistry, hydrogen, containment dose rate, and core exit thermocouple data to determine a realistic estimate of the extent of core damage. The staff has reviewed this procedure and finds that it meets the provisions of Criterion 6 of NUREG-0737, Item II.B.3, and is, therefore, acceptable.

The applicant will test the PASS every 6 months by obtaining a reactor coolant sample through the PASS and comparing the results with a concurrent reactor coolant system sample obtained at the normal sampling station. At the same time, on-line instrumentation will be calibrated and tested. Following system installation, PASS operation will be included as part of the emergency exercises. The staff finds that the provisions of Criterion 9 of NUREG-0737, Item II.B.3, have been met, and the PASS operation is, therefore, acceptable.

The applicant provided a summary of the postaccident sampling analytical procedures and on-line instruments which included the type of analysis, equipment, suitability, range, and analytical method. Suitability was determined through testing using the standard chemical test matrix or by testing in a similar environment. The staff has reviewed the applicant's analytical procedure suitability evaluation and finds it meets the requirements of Criterion 11 of NUREG-0737, Item III.B.3, and is, therefore, acceptable.

By letter dated January 18, 1984, the applicant requested the following:

(1) In Criteria 1b and 1g, change "performing boron analysis from in-line monitoring" to "performing boron analysis by grab samples." Reactor coolant boron analysis will be performed by plasma spectroscopy on a 1000:1 diluted grab sample. This method has suitable sensitivity with diluted samples and can be done in the required time frame. These provisions meet the requirements of Item II.B.3 of NUREG-0737, and are, therefore, acceptable.

(2) In Criterion 5, change the statement that "all valves which are inaccessible for repairs after an accident are environmentally qualified for operation as containment isolation valves and are capable of being opened with a reliable power supply in the event of a loss of offsite power" to "all PASS valves which are inaccessible for repairs after an accident are environmentally qualified for the conditions in which they need to operate." The change reflects the original license condition which will ensure operation of the PASS in the event of an accident. This provision meets the requirements of Item II.B.3 of NUREG-0737, and is, therefore, acceptable.

On the basis of the above evaluation, the staff concludes that the postaccident sampling system meets all the requirements of Item II.B.3 of NUREG-0737 and is, therefore, acceptable.

II.F.1 Attachment 1, Noble Gas Effluent Monitor

In the SER, the staff indicated that the applicant's response to the requirements of Item II.F.1, Attachment 1, did not conform to NUREG-0737 in the following areas:

(1) The applicant had not addressed the calibration of the monitors, calibration frequency and technique, energy dependence of response, monitoring locations or points of sampling, and method of recording the data for the noble gas accident monitors.

(2) The applicant had not addressed the range of sensitivity, energy dependence of response, calibration frequency and technique, vendor's model number, and location of instrument readouts for the main steamline monitors.

(3) The applicant had proposed an unacceptable method to the staff for calculating releases from the safety/relief valves.

(4) The applicant had not provided, for review, the final design description of the as-built system, including piping and instrument drawings, and either a description of procedures for system operation and calibration or copies of procedures for system operation and calibration.

Subsequent to the publishing of the SER, the applicant has submitted additional information on the noble gas effluent and main steamline monitors.

The noble gas monitors will be located on the plant vent stack, the condenser vacuum pump effluent, and the fuel handling building emergency exhaust. The noble gas monitors will be calibrated every refueling outage. These monitors will be field calibrated with a single calibration source and will be calibrated at a single point on the detector's calibration curve. The vendor for these monitors, General Atomic, will provide a primary calibration report. The monitors' response has been provided in Table 1.9-4 of the FSAR. The monitor, located in the plant vent and fuel handling emergency exhaust, will take an isokinetic sample inside the duct of the appropriate release point. For high-activity conditions, another isokinetic nozzle can be used to draw a sample from the sample stream coming from the duct-mounted nozzle. A microprocessor will perform flow control, valve actuation, engineering conversions and other calculations, and control functions in addition to data storage.

The main steamline monitors will consist of one collimated Geiger-Muller tube per steamline. The monitors are mounted within a 3-in.-thick lead shield with a window at the front of the detector. The monitors will be calibrated at the same frequency as the noble gas effluent monitors. The vendor of the main steamline monitors, General Atomic, will be required to provide a primary calibration report on the monitor that includes the response of the detector. These monitors will also be field calibrated with a single calibration source at a single point on the detector's calibration curve. These monitors will have a range of 10^{-1} to 10^{3} μCi/cc of Xe-133 in the main steamline. The energy dependence of the monitor was provided in Table 1.9-4 of the FSAR. Conversion factors have been developed for the main steamline monitors in terms of mR/hr per μCi/cc of pressurized steam. The methodology used to determine the conversion factors is taken from the Reactor Shielding Design Manual by T. Rockwell III. This model accounts for the thickness of the main steamline wall. Readout of all monitor items will be available from the radiation monitoring system computer remote console cathode ray tube (CP-6) and from separate control room readouts found in CP-52 located in the control room. The microprocessor will record release concentrations.

The applicant proposed that the release from the safety/relief valves and the atmospheric steam dump valves be calculated by determination of the concentrations in the main steamline from a calculated conservative conversion factor and from the mass of steam released through the valves. The applicant has indicated that the mass of steam released will be determined by estimating the time each valve is open using the main steam flow recorder, the atmospheric dump valve position, and the setpoint indication.

The staff has reviewed the applicant's proposed method for complying with Item II.F.1, Attachment 1. For both noble gas and main steamline monitors, the energy dependence of the monitor's response will be contained in the calibration report that will be available for review by NRC. The calibration technique will also be available from this report. The applicant's proposed method for complying with the remainder of the requirements for noble gas and main steamline monitors is acceptable to the staff.

The applicant has provided, for review, the final design description of the as-built system in Amendment 31 to the FSAR. This system was reviewed by regional inspectors who determined that the noble gas effluent monitors satisfy the extended and normal range measurement criteria of NUREG-0737 and RG 1.97. Operating and calibration procedures and operator training will be verified by regional inspectors. Implementation of the requirement is not necessary before low-power operation because only small quantities of radionuclide inventory will exist in the reactor coolant system and, therefore, will not affect the health and safety of the public.

II.F.2 Instrumentation for Detection of Inadequate Core Cooling

Supplement 5 stated that the following two open items have to be resolved before an operating license is issued:

(1) The response to Item (2) of II.F.2, Attachment 1 (a primary operator display) is incomplete. It should be clarified.

(2) The response to Item (4) of II.F.2 (documentation required) is not complete. It should include each subsystem of the final inadequate core cooling instrumentation system.

In response to the staff findings in Supplement 5, the applicant has provided additional information in a letter dated October 31, 1983, and FSAR Amendment 34 dated January 13, 1984.

The applicant has clarified the use of the qualified safety parameter display system (QSPDS) for primary and backup inadequate core cooling (ICC) display in the Waterford 3 control room. The QSPDS performs safety-grade signal processing and display of the ICC parameters and is located on the main control panel for reactor protection to facilitate use by the operator. The QSPDS accepts sensor inputs, processes the signals, and transmits the output to its own alphanumeric display and to the plant computer through which the line printer is accessible. All non-Class 1E inputs and interface with the plant computer are isolated from the Class 1E QSPDS equipment.

A spatially oriented core exit thermocouple (CET) temperature map is available on demand from each train of the QSPDS (primary and backup), providing a uniform representative picture of core exit temperature obtained by utilizing 28 CETs (7 per quadrant) dedicated only to that train. A strip chart recorder is provided to allow trending of representative CET temperature for the primary display (QSPDS train A).

Direct readout and hard copy capability is provided for all thermocouple temperatures (direct readout for the 28 CETs associated with each train of the QSPDS can be obtained from the display associated with that train; hard copy capability is via the line printer). Selective readings of core exit temperature, continuous on demand, are available from both the primary and backup displays. On the basis of its review, the staff has found that the applicant's clarification is acceptable.

The applicant has also provided the information in response to Item (4) of the documentation required by NUREG-0737, Item II.F.2, including an evaluation of the conformance of the ICC instrument system to Item II.F.2, Attachment 1, and Appendix B of NUREG-0737. The staff has reviewed and found it in compliance with the requirements.

On the basis of its review, the staff concludes that the applicant's response to the open items stated above is acceptable for the issuance of an operating license.

APPENDIX A

CONTINUATION OF CHRONOLOGY OF SAFETY REVIEW

June 7, 1983	Letter from applicant transmitting revised description of QA program, FSAR Chapter 17.
June 8, 1983	Letter from applicant providing summary of preoperational piping thermal expansion testing.
June 16, 1983	Letter from applicant regarding submittal of information on depressurization and decay heat removal.
June 17, 1983	Letter to applicant requesting test reports reviewed during site audit on environmental qualification program.
June 20, 1983	Letter from applicant transmitting 22nd monthly staffing report.
June 24, 1983	Letter to applicant advising of acceptability of location of backup and primary emergency operating facility.
June 27, 1983	Letter from applicant forwarding program description for resolving TMI Action Plan Item II.K.3.5, "Automatic Trip of Reactor Coolant Pumps."
June 27, 1983	Letter to applicant requesting additional information/ clarification regarding Union Carbide plant explosion in December 1982.
June 29, 1983	Letter from applicant transmitting: (1) "Depressurization and Decay Heat Removal, Response to NRC Questions," CEN-239, June 1983; (2) "Probabilistic Risk Assessment of the Effects of PORVs on Depressurization and Decay Heat Removal," CEN-239, Supplement 2, June 1983; (3) "Depressurization and Decay Heat Removal, Waterford 3 Response to NRC Questions 6a, 6b, 12, 13a, 13c, 13d," June 1983.
June 29, 1983	Letter from applicant regarding core protection calculator addressable constant changes.
June 29, 1983	Letter from applicant transmitting information (including proprietary material) from Shell Oil regarding hazardous materials affecting Waterford.
June 30, 1983	Letter from applicant transmitting Amendment 32 to FSAR.

July 1, 1983	Letter from applicant providing formal notification of deferral of plant-specific simulator - should now be operational by December 31, 1987.
July 5, 1983	Generic Letter 83-26 - Clarification of Surveillance Requirements for Diesel Fuel Impurity Tests.
July 5, 1983	Letter from applicant advising of revised schedule for completion of responses on emergency response capability regarding safety parameter display system.
July 6, 1983	Letter from applicant forwarding report on implementation of Regulatory Guide (RG) 1.97, Revision 2.
July 6, 1983	Generic Letter 83-27 - Surveillance Intervals in Standard Technical Specifications.
July 8, 1983	Issuance of Supplement No. 5 to Safety Evaluation Report.
July 8, 1983	Generic Letter 83-28 - Required Actions Based on Generic Implications of Salem ATWS Event.
July 19, 1983	Letter from applicant transmitting summary technical report on reactor containment building integrated leak rate test.
July 21, 1983	Letter from applicant regarding secondary chemistry.
July 21, 1983	Letter from applicant regarding monitoring of the nuclear plant island structure (NPIS) settlement.
July 21, 1983	Letter from applicant regarding postaccident sampling system.
July 21, 1983	Letter from applicant regarding engineered safety features actuation system surveillance requirements.
July 21, 1983	Letter from applicant regarding licensee event report procedures.
July 21, 1983	Generic Letter 83-30 - Deletion of Standard Technical Specification Surveillance Requirement 4.8.1.1.2.d.6 for Diesel Generator Testing.
July 21, 1983	Letter from applicant regarding closing of issue on RG 1.141, Revision 2 (essential versus nonessential system).
July 21, 1983	Letter from applicant regarding emergency core cooling system preoperational test.
July 25, 1983	Letter from applicant advising of delay in submittal of final report on preservice inspection program.

July 25, 1983	Letter from applicant regarding waste management system.
July 26, 1983	Board Notification 83-105 - Resolution of Differing Professional Opinion Concerning USI A-17 - Systems Interaction Program.
July 28, 1983	Letter from applicant in response to June 27 letter regarding the Union Carbide plant explosion.
July 29, 1983	Generic Letter 83-23 - Safety Evaluation of "Emergency Procedure Guidelines."
August 3, 1983	Letter from applicant advising of program for toxic chemical surveys.
August 3, 1983	Letter from applicant forwarding information on environmental qualification.
August 3, 1983	Letter from applicant advising of procedure for prediction of core damage.
August 5, 1983	Letter from applicant concerning the temporary containment building crane.
August 5, 1983	Letter from applicant concerning environmental qualification of Rosemont transmitters.
August 8, 1983	Letter from applicant forwarding Change 1 to Revision 1 to "Offsite Dose Assessment Manual" and Revision 1 to "Radiological Field Monitoring."
August 8, 1983	Letter from applicant regarding fire water system connection (tertiary backup).
August 9, 1983	Letter from applicant regarding reorganization.
August 11, 1983	Letter from applicant transmitting report of eddy current examination of the steam generator tubes.
August 12, 1983	Letter from applicant regarding fuel assembly response to seismic and LOCA loading, transmitting "Final Assessment of Waterford 3 Fuel Structural Integrity Under Faulted Conditions," CEN-159(c), Revision 1 (proprietary and nonproprietary versions).
August 17, 1983	Letter from Torrey Pines regarding classification of stresses by Combustion Engineering in Potential Finding Report 2448-011.
August 18, 1983	Letter to applicant transmitting Technical Evaluation Report on control of heavy loads.

August 22, 1983	Board Notification 83-126 - Resolution of Differing Professional Opinion Concerning USI A-17 - Systems Interaction Program.
August 22, 1983	Letter from applicant transmitting security plan information.
August 23, 1983	Letter from applicant transmitting revised seismic qualification completion schedule.
August 25, 1983	Board Notification 83-1[illegible]3 - Materials Supplied to Nuclear Industry Companies by Ray Miller Inc. and Tube-Line Corporation.
August 26, 1983	Letter to applicant transmitting request for additional information on emergency planning regarding offsite explosions/fires.
August 30, 1983	Letter from applicant requesting extension to 60-day period for deferring response to Generic Letter 83-28.
August 31, 1983	Board Notification 83-128 - Draft Test Report on Qualification Test Program of Class 1E Solenoid Valves.
September 9, 1983	Letter from applicant requesting exemption from commitment to install fire damper in fire barrier (fire areas 25 and 32).
September 14, 1983	Letter from applicant with revised schedule for completion of responses regarding emergency response requirements.
September 15, 1983	Board Notification 83-133 - Inspection and Enforcement Inquiry Team Report on Allegations.
September 16, 1983	Letter from applicant forwarding response to questions concerning containment purge valves.
September 20, 1983	Letter from applicant regarding depressurization and decay heat removal.
September 20, 1983	Letter from applicant forwarding "Probabilistic Risk Assessment of Effects of PORVs on Depressurization and Decay Heat Removal," CEN-239, Supplement 2, Amendment 1.
September 28, 1983	Letter from applicant transmitting Amendment 33 to FSAR.
September 29, 1983	Letter from applicant in response to August 26 letter regarding emergency planning for offsite explosions/fires.
September 29, 1983	Board Notification 83-144 - Staff Evaluation of Need for PORVs on CE Plants.
September 29, 1983	Letter from applicant transmitting amended implementation schedule regarding requirements for safety parameter display system.

September 29, 1983	Letter from applicant requesting extension of construction completion date to September 30, 1984.
September 30, 1983	Letter from applicant advising of authorized signatories.
October 3, 1983	Letter from applicant advising of installation of removable section in south wall of diesel generator 3BS room.
October 4, 1983	Letter to applicant transmitting Technical Evaluation Reports for site audit of seismic and dynamic qualification program for mechanical and electric equipment.
October 5, 1983	Letter from applicant advising that he will participate with Institute of Nuclear Power Operations and CE Owners Group in responding to Generic Letter 83-28.
October 5, 1983	Letter from applicant regarding turbine missile issue.
October 6, 1983	Board Notification 83-128A - Draft Test Report on Qualification Test Program of Class 1E Solenoid Valves.
October 17, 1983	Letter to applicant transmitting request for additional information concerning concrete mat cracking and water seepage issues.
October 18, 1983	Letter to applicant denying August 30 request.
October 18, 1983	Letter to applicant providing clarification of required actions based on generic implications of Salem anticipated transient without scram events (discussed in Generic Letter 83-28).
October 19, 1983	Generic Letter 83-33 - NRC Positions on Certain Requirements of Appendix R to 10 CFR 50.
October 20, 1983	Letter from applicant stating that failure of core protection calculator software to apply failed control element assembly calculator penalty factor to local power density has been corrected.
October 20, 1983	Letter from applicant transmitting Harstead Engineering Associates report, "Analysis of Cracks and Water Seepage in Foundation Mat."
October 24, 1983	Letter from applicant transmitting information on Nuclear Operations Organization.
October 24, 1983	Letter from applicant forwarding revised Emergency Plan procedures regarding contaminated, injured, or ill personnel.
October 26, 1983	Meeting with applicant to discuss safety significance of cracks and water seepage in foundation base mat.

October 27, 1983	Letter from applicant regarding depressurization and decay heat removal.
October 31, 1983	Generic Letter 83-38 - NUREG-0965, "NRC Inventory of Dams."
October 31, 1983	Letter from applicant concerning inadequate core cooling instrumentation.
October 31, 1983	Letter from applicant in response to basemat questions.
November 2, 1983	Generic Letter 83-35 - Clarification of TMI Action Plan Item II.K.3.31.
November 4, 1983	Letter from applicant forwarding response to Generic Letter 83-28.
November 7, 1983	Letter from applicant forwarding information concerning non-safety and safety-related electrical equipment and post-accident monitoring equipment.
November 9, 1983	Letter from applicant regarding toxic chemical surveys.
November 15, 1983	Meeting with applicant to discuss emergency feedwater system control design.
November 16, 1983	Letter from EBASCO transmitting sketch that is basis for FSAR Figure 2.5-118.
November 17, 1983	Letter from applicant transmitting water chemistry report for ground water sample associated with basemat.
November 18, 1983	Letter to applicant forwarding request for additional information on emergency feedwater system.
November 29, 1983	Letter from applicant forwarding information on postaccident chemistry procedures and on-line instrumentation.
November 30, 1983	Letter from applicant advising of proposed construction of process steamline.
December 2, 1983	Generic Letter 83-32 - Staff Recommendations Regarding Operator Action for Reactor Trip and Anticipated Transients Without Scram.
December 7, 1983	Letter to applicant regarding process steam pipeline between Waterford 1 and 2 and Union Carbide plant.
December 12-13, 1983	Site visit concerning physical protection.
December 15, 1983	Letter from applicant forwarding response to Facility Staffing Survey.

December 16, 1983	Letter from applicant transmitting revision to training and qualification plan (security program).
December 19, 1983	Generic Letter 83-42 - Clarification to Generic Letter 81-07 Regarding Response to NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants."
December 19, 1983	Letter from applicant forwarding "Procedures Generation Package," Volumes 1-5.
December 19, 1983	Generic Letter 83-43 - Reporting Requirements of 10 CFR Part 50, Sections 50.72 and 50.73 and Standard Technical Specifications.
December 19, 1983	Letter from applicant forwarding "Detailed Report on Evacuation of 821211" regarding St. John and St. Charles Parishes.
December 20, 1983	Generic Letter 83-44 - Availability of NUREG-1021, "Operator Licensing Examiner Standards."
December 23, 1983	Letter from applicant concerning emergency feedwater control system.
December 29, 1983	Letter from applicant regarding redesign of component cooling water system (concerning Generic Letter 83-10A).
January 5, 1984	Board Notification 84-004 - Environmental Qualification Briefing of Chairman by Sandia.
January 5, 1984	Generic Letter 84-01 - NRC Use of the Terms "Important to Safety" and "Safety Related."
January 6, 1984	Generic Letter 84-02 - Notice of Meeting Regarding Facility Staffing.
January 12, 1984	Letter from applicant transmitting Revision 3 to Safeguards Contingency Plan.
January 13, 1984	Letter from applicant transmitting Amendment 34 to FSAR.
January 13, 1984	Generic Letter 84-03 - Availability of NUREG-0933 on Generic Safety Issues.
January 17, 1984	Meeting with applicant to discuss emergency feedwater system control design.
January 18, 1984	Board Notification 84-011 - NRC Use of Terms "Important to Safety" and "Safety Related."
January 18, 1984	Letter from applicant regarding postaccident sampling system.
January 25, 1984	Letter from applicant transmitting safety information booklet.

January 26, 1984	Meeting with applicant to discuss availability of component cooling water to reactor coolant pumps.
January 26, 1984	Letter from applicant forwarding description of actions and procedural requirements related to leak reduction program.
January 30, 1984	Letter from applicant forwarding list of safety-related mechanical equipment.
January 30, 1984	Letter from applicant transmitting Emergency Plan, Revision 6.
February 1, 1984	Letter to applicant transmitting request for additional information on core bypass flow.
February 1, 1984	Generic Letter 84-04 - Safety Evaluation of Westinghouse Topical Reports Dealing With Elimination of Postulated Pipe Breaks in PWR Primary Main Loops.
February 6, 1984	Letter from applicant forwarding information in response to Generic Letter 83-28.
February 6, 1984	Letter from applicant advising that response to Item 4.5.1 of Generic Letter 83-28 has been provided in "Manual Reactor Trip Test."
February 7, 1984	Letter from applicant regarding toxic chemical surveys.
February 9, 1984	Letter from applicant regarding control of heavy loads - special lifting devices.
February 10, 1984	Letter from applicant forwarding relief requests resulting from preservice inspection hanger/hanger support visual examinations.
February 10, 1984	Letter from applicant regarding environmental qualification.
February 13, 1984	Board Notification 84-032 - Additional Information on Environmental Qualification.
February 16, 1984	Letter from licensee regarding licensee qualification.
February 16, 1984	Letter from applicant requesting disclosure of allegations information.
February 16, 1984	Board Notification 84-030 - Combustion Engineering Auxiliary Pressurizer Spray Systems.
February 16, 1984	Letter from applicant in response to Generic Letter 83-28, Item 4.1, "Reactor Trip System Reliability."
February 16, 1984	Letter from applicant forwarding information on changes to component cooling water system to enhance availability of cooling to reactor coolant pump seals.

February 16, 1984	Board Notification 84-33 - Task Action Plan for USI A-17.
February 16, 1984	Letter from applicant regarding emergency feedwater control system.
February 20, 1984	Letter to applicant regarding deletion of home telephone numbers, unlisted utility numbers, etc. from emergency plans.
February 21, 1984	Letter from applicant forwarding documents on radioactive waste solidification process control programs (proprietary).
February 22, 1984	Letter from applicant forwarding information on core bypass flow (proprietary and nonproprietary versions).
February 23, 1984	Letter from applicant forwarding Seismic Qualification Review Team (SQRT) comments from audit report.
February 23, 1984	Letter from applicant forwarding information on fire protection training program.
February 27, 1984	Letter from applicant regarding loss of non-Class 1E instrumentation and control power system bus during operation.
February 28, 1984	Letter from applicant forwarding information for control room design review.
February 29, 1984	Letter from applicant transmitting "An Evaluation of the Natural Circulation Cooldown Test Performed at the San Onofre Nuclear Generating Station," CEN-259.
March 1, 1984	Letter from applicant supplementing August 5, 1983, letter regarding Rosemount transmitters.
March 1, 1984	Letter to applicant summarizing January 26, 1984, meeting regarding experience of operators.
March 2, 1984	Letter from applicant transmitting postaccident sampling system status report.
March 2, 1984	Letter from applicant regarding core protection calculator rod bow penalty factors.
March 2, 1984	Letter from applicant confirming that all SQRT files are closed.
March 5, 1984	Letter from applicant regarding containment pressure setpoint.
March 5, 1984	Letter from applicant in response to Generic Letter 83-28, providing information on post-maintenance testing of reactor trip system components.

March 5, 1984	Letter from applicant forwarding information in response to Generic Letter 81-04.
March 8, 1984	Letter from applicant forwarding information related to previous submittal of Emergency Plan information.
March 8, 1984	Letter from applicant forwarding letters of agreement from chemical industries within 5 mi of Waterford.
March 8, 1984	Letter from applicant forwarding information in response to Generic Letter 83-28, Item 4.5.3.
March 8, 1984	Letter from applicant regarding activities and commitments to be completed before fuel loading.
March 8, 1984	Letter from applicant forwarding information in response to Generic Letter 83-28, Item 4.2.
March 12, 1984	Letter from applicant regarding containment pressure setpoint.
March 12, 1984	Letter from applicant advising that commitment for vendor review of emergency operating procedures fulfilled.
March 13, 1984	Letter from applicant providing justification for interim operation pending complete environmental qualification of ex-core neutron flux detectors.
March 14, 1984	Board Notification 84-050 - Environmental Qualification: Commission Policy Statement and Proposed Rulemaking.
March 15, 1984	Letter from applicant regarding training program.
March 15, 1984	Meeting with applicant to hold final discussions on Technical Specifications before licensing.
March 15, 1984	Letter from applicant forwarding information on revised corporate command center and emergency news center (proprietary and nonproprietary versions).
March 16, 1984	Letter from applicant providing justification for exemption from type C leak testing, list of isolation valves within essential system, and penetration isometric drawings.
March 20, 1984	Letter from applicant regarding activities and commitments to be completed before fuel loading (replaces March 8 letter).
March 22, 1984	Letter from applicant forwarding completed operating shift experience forms.
March 26, 1984	Meeting with applicant to discuss basemat adequacy.

March 26, 1984	Letter from applicant requesting review of enclosed exceptions to 10 CFR 50, Appendix R.
March 27, 1984	Letter from applicant forwarding information on leak testing of engineered safety features heating, ventilation, and air conditioning ductwork and housings.
March 27, 1984	Letter from applicant regarding licensee qualification training.
March 28, 1984	Letter from applicant forwarding Combustion Engineering (CE) Shop Pre-Service Inspection Report.
March 28, 1984	Letter from applicant regarding environmental qualification.
March 28, 1984	Letter from applicant forwarding "Alert/Notification System."
March 28, 1984	Letter from applicant requesting that proposed Technical Specifications be revised to reflect monthly (31-day) turbine valve cycling frequency.
March 28, 1984	Letter from applicant forwarding CE Shop Inspection Report.
March 29, 1984	Letter from applicant regarding four human engineering deficiencies and corrective actions identified during control room design review.
March 30, 1984	Letter to applicant requesting additional information regarding auxiliary pressurizer spray systems.
April 2, 1984	Generic Letter 84-05 - Change to NUREG-1021, "Operator Licensing Examiner Standards."
April 2, 1984	Letter to applicant requesting response to enclosed technical areas.
April 4, 1984	Generic Letter 84-08 - Interim Procedures for NRC Management of Plant-Specific Backfitting.
April 6, 1984	Letter to applicant regarding Federal Emergency Management Agency's review of revised public information brochure.
April 9, 1984	Letter from applicant forwarding justification for interim operation for Borg-Worner actuators.
April 10, 1984	Meeting with applicant to discuss appeal of requirements for Appendix J type C testing of nine additional penetrations.
April 10, 1984	Letter from applicant regarding detailed control room design review program.

April 10, 1984	Letter from applicant forwarding "Environmental Qualification of Waste Gas Compressor A, B," "Environmental Qualification of Steam Generator Hydraulic Snubber SG-MSNB-734-1A," and "Environmental Qualification of 2BM-F108 A/B Valve."
April 11, 1984	Letter from applicant forwarding Revision 8 to Security Plan.
April 12, 1984	Letter from applicant regarding potential single-failure vulnerability of the auxiliary pressurizer spray.
April 12, 1984	Letter from applicant regarding natural circulation/boron mixing test.
April 13, 1984	Letter from applicant providing updated information on postaccident sampling system.
April 16, 1984	Letter from applicant forwarding information on construction adequacy of basemat.
April 16, 1984	Letter from applicant forwarding "Safety Parameter Display System."
April 17, 1984	Letter from applicant providing license condition commitment regarding fuel rod pressure.
April 17, 1984	Letter from applicant providing commitment regarding fission gas release analysis.
April 17, 1984	Letter from applicant advising that response to allegations will be provided by April 27.
April 18, 1984	Letter to applicant forwarding request for additional information on shutdown cooling system relief valves.
April 19, 1984	Letter from applicant regarding modifications to circulating water system intake structure.
April 19, 1984	Letter from applicant forwarding calculation for containment pressure trip setpoint.
April 19, 1984	Letter from applicant regarding Appendix J type C leak testing meeting held April 10.
April 19, 1984	Letter to applicant forwarding draft Technical Specifications and requesting certification that the draft reflects the plant.
April 23, 1984	Letter from applicant supplementing April 17 letter regarding fission gas release analysis.
April 25, 1984	Letter from applicant forwarding additional information concerning radioactive waste solidification process control program.

April 25, 1984	Letter from applicant advising of plant readiness for fuel loading by May 30, 1984.
April 26, 1984	Generic Letter 84-10 - Administration of Operating Tests Prior to Initial Criticality.
April 26, 1984	Meeting with applicant to close out remaining items under review (including emergency feedwater control system) by Instrumentation and Control Systems Branch.
April 26, 1984	Letter from applicant transmitting Amendment 35 to FSAR.
April 26, 1984	Letter from applicant regarding authorized staffing levels.
April 27, 1984	Letter from applicant forwarding responses to allegations.
April 27, 1984	Letter from applicant regarding shutdown cooling system relief valves.
April 30, 1984	Generic Letter 84-12 - Compliance With 10 CFR Part 61 and Implementation of the Radiological Effluent Technical Specifications and Attendant Process Control Program.
April 30, 1984	Letter from applicant forwarding information on licensee qualification.
May 2, 1984	Letter from applicant regarding April 26 meeting concerning emergency feedwater control system.
May 2, 1984	Letter from applicant forwarding information on procedures generation package.
May 3, 1984	Letter from applicant regarding heating, ventilation, and air conditioning testing.
May 3, 1984	Letter from applicant regarding backup route alerting in areas where sirens are to be installed.
May 3, 1984	Generic Letter 84-13 - Technical Specifications for Snubbers.
May 4, 1984	Letter from applicant forwarding corrected Attachment II for May 2 letter.
May 7, 1984	Letter from applicant regarding submittal of information on detailed control room design review program plan.
May 7, 1984	Letter to applicant forwarding request for additional information on procedures generation package.
May 7, 1984	Letter from applicant forwarding Revision 7 to Emergency Plan and related information.

May 3, 1984	Letter from applicant regarding use of main steam flow recorders.
May 8, 1984	Generic Letter 84-09 - Recombiner Capability Requirements of 10 CFR 50.44(C)(3)(ii).
May 9, 1984	Meeting with applicant to discuss Reactor Systems Branch comments on Technical Specifications.
May 10, 1984	Letter from applicant forwarding response to radwaste concerns.
May 10, 1984	Letter from applicant forwarding updated drawings for emergency feedwater actuation system.
May 11, 1984	Generic Letter 84-14 - Requalification Training Program.
May 11, 1984	Letter from applicant in response to Generic Letter 83-28.
May 14, 1984	Meeting with applicant to discuss Reactor Systems Branch comments on Technical Specifications.
May 14, 1984	Letter from applicant forwarding responses to requests for additional information on NPIS basemat.
May 14, 1984	Letter from applicant regarding engineered safety features actuation system subgroup relay testing, as discussed in meeting of April 26, 1984.
May 14, 1984	Letter from applicant providing résumés for control room and shift supervisors.
May 14, 1984	Letter to applicant forwarding request for additional information on financial qualifications.
May 15, 1984	Letter from applicant forwarding 1983 Annual Report.
May 15, 1984	Letter from applicant forwarding revised emergency news center and corporate command center instructions.
May 16, 1984	Letter from applicant transmitting "Nuclear Plant Island Structure Wall Hairline Cracks Evaluation."
May 17, 1984	Letter from applicant transmitting Amendment 36 to FSAR.
May 17, 1984	Letter from applicant forwarding resolution of 10 CFR 50, Appendix R, audit findings.
May 18, 1984	Letter to applicant forwarding request for additional information on Technical Specifications.
May 18, 1984	Letter from applicant forwarding information on modifications to the circulating water system intake structure.

May 18, 1984	Letter from applicant supplementing February 26, 1984, request for relief from 10 CFR 50, Appendix R.
May 21, 1984	Management meeting regarding plant readiness for operation.
May 23, 1984	Letter from applicant transmitting clarifications to Chapter 14 of FSAR.
May 23, 1984	Letter from applicant stating commitment to perform confirmatory tests to verify presence of Boraflex in spent fuel storage racks within 9 months after fuel loading.
May 25, 1984	Letter from applicant advising that financial information will be provided by June 1.
May 29, 1984	Letter to applicant regarding operator shift staffing.
May 29, 1984	Letter from applicant forwarding response to questions on Technical Specifications.
May 29, 1984	Letter from applicant forwarding response to request for financial information.
May 30, 1984	Letter to applicant transmitting request for additional information on procedures generation package.
May 30, 1984	Letter to applicant regarding Technical Specifications.

APPENDIX B

BIBLIOGRAPHY

Atomic Safety Licensing Board," Memorandum and Order, "Regarding Applicant's Motion for Reconsideration or Clarification," Dec. 14, 1982.

---, Partial Initial Decision of November 3, 1982.

Code of Federal Regulations, Title 10, "Energy" (including General Design Criteria), U.S. General Services Administration, Office of the Federal Register, National Archives and Records Service, U.S. Government Printing Office, Washington, D.C.

Combustion Engineering Inc. (CE), Topical Report CEN-39(A)-P, Rev. 2, "CPC Protection Algorithm Software Change Procedure," Arkansas Nuclear One, Unit 2, Docket No. 50-368, Dec. 21, 1978.

---, Topical Report CEN-159(C)-P, Rev. 1-P, "Final Assessment of Waterford-3 Structural Integrity Under Faulted Conditions," July 15, 1983.

---, Topical Report CEN-259, "An Evaluation of the Natural Circulation Cooldown Tests Performed At the San Onofre Nuclear Generating Station Compliance With the Test Requirements of Branch Technical Position RSB 5-1," Jan. 1984.

---, Topical Report CENPD-178, Rev. 1, "Structural Analysis of Fuel Assemblies for Seismic and Loss of Coolant Accident Loading," Aug. 1981.

---, Topical Report CENPD-225, "Fuel and Poison Rod Bowing," Oct. 1976.

---, TR-ESS-037, "Shop Preoperation Inspection Program."

Institute of Nuclear Power Operations, Document GPG-01, Rev. 1, "Nuclear Power Plant Shift Technical Advisor."

Instrument Society of America (ISA), "Effect of Fluid Compressibility on Torque in Butterfly Valves," by F. T. Harthun, ISA Transactions, Vol. 8, No. 4, 1969.

Louisiana Power & Light Company, "Final Safety Analysis Report, Waterford 3 Steam Electric Station, Unit No. 3."

Quarantelli, E. L., "Evacuation Behavior: Case Study of the Taft Louisiana Chemical Tank Explosion Incident," May 1983.

Rockwell, T., III, Reactor Shielding Design Manual, 1956.

Torrey Pines Technology, Technical Report GA-C16900, "Independent Design Review of Waterford Steam Electric Station Unit 3 Emergency Feedwater System," Mar. 1983.

U.S. Nuclear Regulatory Commission, NUREG-0582, "Water Hammer in Nuclear Power Plants," Apr. 1979.

---, NUREG-0612, "Control of Heavy Loads at Nuclear Plants," 1980.

---, NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," Feb. 1980; Rev. 1, Nov. 1980.

---, NUREG-0696, "Functional Criteria for Emergency Response Facilities," Feb. 1981.

---, NUREG-0712, "Safety Evaluation Report Related to the Operation of San Onofre Nuclear Generating Station, Units 2 and 3," Feb. 1981; Supplement 1, Feb. 1981.

---, NUREG-0737, "Clarification of TMI Action Plan Requirements," Nov. 1980; Supplement 1 (Generic Letter 82-33), Dec. 1982.

---, NUREG-0787, "Safety Evaluation Report Related to the Operation of Waterford Steam Electric Station, Unit No. 3," July 1981; Supplement 1, Oct. 1981; Supplement 2, Jan. 1982; Supplement 3, Apr. 1982; Supplement 4, Oct. 1982; Supplement 5, June 1983.

---, NUREG-0800 (formerly NUREG-75/087), "Standard Review Plan for Review of Safety Analysis Reports for Nuclear Power Plants--LWR Edition" (including Branch Technical Positions), July 1981.

---, NUREG-1044, "Evaluation of the Need for a Rapid Depressurization Capability for CE Plants," Draft Mar. 1984.

U.S. Regulatory Commission, Office of Inspection and Enforcement, IE Bulletin 79-27, "Loss of Non-Class 1E Instrumentation and Control Power System Bus During Operation," Nov. 30, 1979.

---, IE Circular 78-02, "Proper Lubricating Oil for Terry Turbines," Apr. 20, 1978.

---, IE Report No. 50-382/83-08, "Regarding Notice of NRC Emergency Preparedness Appraisal for Waterford 3," Feb. 3, 1983.

---, IE Report No. 50-382/84-02, "Regarding Emergency Appraisal Followup," Mar. 29, 1984.

APPENDIX C

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
LETTER DATED OCTOBER 18, 1983

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

October 18, 1983

Honorable Nunzio J. Palladino
Chairman
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Dr. Palladino:

SUBJECT: NEED FOR RAPID DEPRESSURIZATION CAPABILITY IN NEWER COMBUSTION ENGINEERING, INC. PLANTS

During its 282nd meeting, October 13-15, 1983, the Advisory Committee on Reactor Safeguards (ACRS) reviewed analyses of the NRC Staff and of a Combustion Engineering Owners Group (CEOG) regarding the need for addition of power operated relief valves (PORVs) to certain nuclear power plants designed by Combustion Engineering, Inc. (CE). This matter had been reviewed previously by a Subcommittee of the ACRS on October 4, 1983, and earlier on January 27, 1983 and March 16, 1982. PORVs are automatic and remotely operable valves installed on the reactor coolant system (RCS) pressurizer in most PWRs. The valves were originally intended to intercept overpressure challenges to code safety valves. These latter valves are prone to failure to automatically reclose tightly following pressure relieving actuation. The PORVs were perceived to be more manageable in this respect in that they can be closed on demand and can be isolated by a block valve.

Analysis and experience have shown RCS pressure to be more easily controlled than had been recognized earlier so that the need for PORVs in avoiding code safety valve actuation is not now believed to be an important consideration. For that reason, CE, in its most recent plant designs, has not included PORVs in the RCS. Their reasoning is that leakage and the potential for spurious actuation of PORVs (creating, in effect, a small or medium break LOCA) are detrimental to both safety and operating efficiency.

However, within the past few years the PORV has come to be seen as offering other advantages. For one, it is a means to rapidly depressurize the RCS when desired, for example, to minimize leakage to the secondary side following failure of a steam generator tube. A second advantage is as a controlled means to remove steam or hot water from the RCS so that cooler water can be injected by the high pressure safety injection (HPSI) pumps. This is the so-called "feed and bleed" cooling process by which heat can be removed from the RCS and hence the reactor core. Because these advantages must be weighed against the disadvantages mentioned above and the cost of installing PORVs, the NRC Staff and CEOG each have made an extensive analysis of the pros and cons.

The NRC Staff has concluded that the CE plants without PORVs meet all regulatory requirements, with some minor exceptions which can be rather easily corrected. Further, they have concluded that these CE plants, which are equipped with reliable, auxiliary pressurizer sprays (APS) can effect moderate rates of depressurization to accommodate certain transients more effectively than can be done in other PWRs which have PORVs, but which do not have APS. The NRC Staff has also analyzed on a probabilistic basis accidents beyond the design basis accidents, including:

- multiple steam generator tube failures,

- total loss of feedwater,

- small break LOCA without HPSI,

- pressurized thermal shock, and

- ATWS.

The NRC Staff has concluded that addition of PORVs could be advantageous in permitting "feed and bleed" heat removal following loss of all feedwater, and that there would be some advantage in having PORVs provide additional pressure relief for ATWS, and in the case of failure of a large number of steam generator tubes. For the other accident sequences, they conclude that PORVs would provide no improvement over existing systems in the CE plants. The NRC Staff's overall cost-benefit analysis concludes there would be a slight advantage in adding PORVs over not adding PORVs. They acknowledge that the advantage is small compared with uncertainty in the analysis. However, the Staff also states it is their judgment that PORVs will provide an additional margin of safety in providing an effective, alternative means for depressurizing the RCS and thus provide greater flexibility in means for emergency core cooling.

Based on this judgment, the NRC Staff has concluded that PORVs should be required to be backfitted to the CE plants in question. However, they have also concluded that implementation of this requirement need not be hurried, and should be integrated with new requirements for decay heat removal systems that evolve from Task Action Plan A-45.

Analysis by the CEOG has produced results similar to those of the NRC Staff. They conclude the plants meet all regulatory requirements with the minor exceptions alluded to above. Their cost-benefit analysis shows a very small disadvantage in adding PORVs. Several differences in assumptions and data used by CEOG and those used by the NRC Staff apparently account for this conclusion, opposite from that of the NRC Staff. These differences have not been resolved. However, as with the cost-benefit analysis by the NRC Staff, the calculated margin is small compared with uncertainties.

Although the CEOG acknowledges that PORVs could provide an emergency means to depressurize the RCS, they have concluded that depressurization by the APS or by rapid secondary side cooldown is much to be preferred. It is their judgment that PORVs should not be added.

The Committee believes there is so nearly a standoff between costs and benefits that extensive efforts to resolve differences or improve assumptions in the analyses are not warranted. A decision to require or not to require addition of PORVs must hinge on largely nonquantitative judgments.

Under some circumstances there might be significant safety advantage in having available an effective backup means to depressurize the RCS. On the other hand, maintaining integrity of the primary pressure boundary and removing heat through systems designed for that purpose, i.e., the steam generators, is generally preferable, even in emergency situations.

The Committee agrees with the NRC Staff's recommendation to integrate any new requirements for rapid depressurization into the more comprehensive new requirements for improvements to decay heat removal systems expected to be forthcoming from Task Action Plan A-45 within one year. We see no need for earlier resolution of the PORV issue.

Sincerely,

J. J. Ray
Chairman

APPENDIX D

REVIEW OF THE PRESERVICE INSPECTION RELIEF REQUESTS
WATERFORD UNIT 3

D.1 INTRODUCTION

This appendix was prepared with the technical assistance of Department of Energy (DOE) contractors from the Battelle Pacific Northwest Laboratories.

For nuclear power facilities whose construction permits were issued on or after January 1, 1974, 10 CFR 50.55(g)(3) specifies that components shall meet the preservice examination requirements set forth in editions and addenda of Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code applied to the construction of the particular component. The provisions of 10 CFR 50.55a(g)(3) also state that the components (including supports) may meet the requirements set forth in subsequent editions of this Code, which are incorporated by reference in 10 CFR 50.55a(b) subject to the limitations and modifications listed therein.

On February 9, 1983, the Louisiana Power & Light Company (LP&L, the applicant) submitted the Waterford 3 preservice inspection (PSI) program for examinations performed at the plant site based on the 1977 Edition of Section XI through the Summer 1978 Addenda of the ASME Code. The visual inspection program is being conducted in accordance with the 1980 Edition of Section XI through the Winter 1980 Addenda. The preservice examination of the welds of the principal components of the reactor coolant pressure boundary, such as the reactor pressure vessel, steam generators, pressurizer, and reactor coolant pump casings, were performed in the Combustion Engineering fabrication shop based on the 1974 Edition of Section XI through the Summer 1974 Addenda.

In letters dated July 25, 1983, and February 10, 1984, the applicant submitted requests for relief from ASME Code requirements and provided supporting information pursuant to 10 CFR 50.55a(a)(2)(i).

As a result of the review of this information, the staff has determined that certain preservice examinations are impractical and performing these required examinations would result in hardships or unusual difficulties without compensating increase in the level of quality and safety. The basis for this conclusion is discussed in the subsequent paragraphs of this appendix.

D.2 TECHNICAL REVIEW CONSIDERATIONS

(1) The construction permit for Waterford 3 was issued on November 14, 1974. The ASME first published rules for inservice inspection in the 1970 Edition of Section XI. No preservice or inservice inspection requirements existed before that date. Because the plant system design and ordering of long lead time components were well under way by the time the Section XI rules became effective, full compliance with the exact Section XI access and inspectability requirements of the Code are not always practical.

(2) Verification of as-built structural integrity of the primary pressure boundary is not dependent on the Section XI preservice examination. The applicable construction codes to which the primary pressure boundary was fabricated contain material, design, fabrication, examination, and testing requirements that by themselves provide the necessary assurance that the components are capable of performing safely under all operating conditions reviewed in the FSAR and described in the plant design specifications. As a part of these examinations, all of the primary pressure boundary full penetration welds were volumetrically inspected (radiographed) and the system was subjected to hydrostatic pressure tests.

(3) The intent of the preservice examination is to establish a reference or baseline before the initial operation of the facility. The results of subsequent inservice examinations can then be compared with the original condition to determine if changes have occurred. If review of the inservice inspection results shows no change from the original condition, no action is required. In the case where baseline data are not available, all indications must be treated as new indications and evaluated accordingly. Section XI of the ASME Code contains acceptance standards that may be used as the basis for evaluating the acceptability of such indications.

(4) Other benefits of the preservice examination include providing redundant or alternative volumetric inspection of the primary pressure boundary using a test method different from that employed during the component fabrication. Successful performance of a preservice examination also demonstrates that the welds can be effectively inspected during the subsequent inservice examination using a similar test method.

In the case of Waterford 3, a large portion of the ASME Code-required preservice examinations was performed. The staff has concluded that failure to perform a 100% preservice examination of the welds identified below will not significantly affect the assurance of the initial structural integrity.

(5) In some cases where the required preservice examinations were not performed to the full extent specified by the applicable ASME Code, the staff will require that these or supplemental examinations be conducted as part of the inservice inspection program. The staff has concluded that requiring these supplemental examinations to be performed at this time (before plant startup) would result in hardships or unusual difficulties without a compensating increase in the level of quality and safety. The performance of supplemental examinations, such as surface examinations, in areas where volumetric inspection is difficult will be more meaningful after a period of operation. Acceptable preoperational integrity has already been established by similar Section III (ASME Code) fabrication examinations.

In cases where parts of the required examination areas cannot be effectively examined because of a combination of component design or current inspection technique limitations, the staff will continue to evaluate the development of new or improved volumetric examination techniques. As improvements in these areas are achieved, the staff will require that these new techniques be made a part of the inservice examination requirements of those components or welds that received a limited preservice examination.

D.3 EVALUATION OF RELIEF REQUESTS

The applicant requested relief from specific preservice inspection requirements and provided supporting information in letters dated July 25, 1983, and February 10, 1984. On the basis of the information submitted by LP&L and the staff's review of the design, geometry, and materials of construction of the components, certain preservice requirements of the ASME Code, Section XI, have been determined to be impractical and, if implemented, would result in hardships or unusual difficulties without a compensating increase in the level of quality and safety. Therefore, pursuant to 10 CFR 50.55(a)(2), the staff's conclusions that these preservice requirements are impractical are justified as follows:

(1) Circumferential and Longitudinal Pipe Welds With Access Limitations (Relief Request #PSI-01)

Code Requirement

Examination Category B-J - A surface examination of the outside diameter (OD) and a volumetric examination of the lower one-third volume of the weld are required for all piping 4-in. nominal pipe size and greater. A surface examination only is required for pipes less than 4 in. pipe size.

Examination Category C-F - Volumetric examination is required for circumferential butt welds and branch connections exceeding 1/2-in. wall thickness including the weld metal and base metal for one wall thickness by a sampling procedure defined in Paragraph IWC-2500. A surface examination is required for piping with wall thickness of 1/2 in. or less.

Code Relief Request

Relief was requested from performing 100% of the Code-required examination.

Reason for Request

The design of Class 1 and Class 2 piping systems has welded joints, such as pipe-to-elbow and pipe-to-component, which physically obstruct all or part of the required Section XI examinations from the elbow or component side of the weld specified.

(a) All partial examinations were due to component configuration or non-removable restraints.

(b) Extensive surface preparation was done to maximize coverage.

(c) Alternative or partial examinations were used wherever feasible.

(d) Ultrasonic test examination coverage for PSI included essentially 100% of the weld required volume (WRV) rather than just the one-third thickness required by the Code.

(e) Essentially 100% of the total number of welds in Class 2 piping systems were examined during the preservice inspection; none were exempted on the basis of multiple streams performing the same function.

Alernative Examinations

(a) Liquid penetrant testing was used in areas inaccessible for magnetic particle testing (MT), where MT was the selected method of examination.

(b) Alternative angles, search units, vee-paths and other techniques were used to provide ultrasonic coverage, where required, to the maximum extent practical.

Staff Evaluation

In the letter dated July 15, 1983, the applicant provided a detailed summary of the ASME Code Class 1 and 2 piping system welds that received a limited or partial examination. The summary report identifies the specific weld, the examination zone and corresponding isometric drawing, the required examination method, the specific cause for the incomplete examination, the region of the weld actually inspected, and alternative examinations. The summary report contains 128 pages of data describing limited examinations of approximately 530 welds. During the review of these data, the staff considered the applicant's examination procedures for both Class 1 and 2 piping, which includes provisions that significantly exceed the ASME Code requirement for the extent of volumetric examination. In addition, the applicant examined essentially 100% of the total number of welds in the Class 2 piping systems and did not use sampling permitted by the ASME Code based on multiple streams performing the same function.

The staff reviewed the documentation submitted by the applicant in the letter dated July 25, 1983, and determined that the applicant has examined the welds to the maximum extent possible. The staff concludes that the limited Section XI examinations, the examinations performed during fabrication, and the hydrostatic test demonstrate an acceptable level of preservice structural integrity.

(2) Pressurizer Surge Line Weld No. 16-012 (Relief Request #PSI-02)

Code Requirement

A surface examination of the OD and a volumetric examination of the lower one-third volume of the weld are required.

Code Relief Request

Relief was requested from performing the entire surface and volumetric examination required by the Code.

Reason for Request

Because of its design, weld 16-012 is midway through a 4-ft-thick concrete wall enclosing the pressurizer. Insufficient clearance exists for inserting examination materials or equipment. A volumetric examination consisting of a radiographic examination was performed during fabrication to meet Section III requirements under weld designation number CE 209-751.

Staff Evaluation

The staff has determined that examination of this weld to the extent required by the Code is impractical. The staff has determined that the radiography performed during fabrication and the hydrostatic test demonstrate an acceptable level of preservice structural integrity.

(3) Pressure-Retaining Class 2 Welds in the Suction Side of the Safety Injection System (Relief Request #PSI-03)

Code Requirement

A surface examination is required for welds 1/2-in. or less nominal pipe thickness. A surface and volumetric examination is required for welds exceeding 1/2-in. nominal pipe thickness.

Code Relief Request

Relief is requested to perform a volumetric examination on a sampling basis instead of the Code-required surface examination.

Alternative Examination Program

A 10% volumetric sampling, using ultrasonic techniques, has been applied to welds in the safety injection suction lines as described in the PSI program submittal.

Reason for Request

(a) Design pressure and temperature are specified as 160 psig and 250°F, respectively, based on accident conditions. However, for all other operating conditions, the pressure and temperature will be approximately equal to ambient. Therefore, the exemption of IWC-1220(a) of the 1974 Edition through Summer 1975 Addenda is cited.

(2) This piping is susceptible to intergranular attack because it normally contains stagnant borated water and is fabricated from Type 304 stainless steel. Therefore, ultrasonic inspection would be more relevant for these welds.

Staff Evaluation

The use of exclusion from examination criteria defined in Paragraph IWC-1220 of Section XI of the ASME Code and the method of examination (defined in Table IWC-2500-1, Category C-F) may result in welds in certain engineered safety features being excluded from volumetric examination. Surface examination methods are generally performed on the ASME Code Class 2 piping welds during construction. The staff has determined that the applicant has proposed an alternative examination program that was more effective for detecting potential subsurface defects; therefore, the staff finds the alternative program acceptable.

(4) Circumferential Butt Welds in Containment Penetrations, Examination Category C-F (Relief Request #PSI-04)

Code Requirement

A volumetric and surface examination is required for piping greater than 1/2-in. nominal pipe thickness.

Reason for Request

(a) The subject welds are totally enclosed in guard pipes and are completely inaccessible. The design pressure and temperature of the guard pipes are equal to the maximum operating pressure and temperature of the enclosed pipes.

(b) An augmented ISI program has been instituted to ensure the structural integrity of high-energy-fluid piping greater than 4-in. nominal pipe thickness penetrating containment.

Alternative Examination Program

The PSI program submittal contains detailed information regarding the Waterford 3 augmented ISI program.

Staff Evaluation

The staff has determined that the preservice volumetric and surface examination of containment penetration welds totally enclosed in guard pipe is impractical. The staff has reached the conclusion that the radiography performed during fabrication demonstrates an acceptable level of preservice structural integrity and the augmented volumetric examination of the high-energy-fluid piping will provide additional assurance of structural integrity.

(5) Pressure-Retaining Welds in Shutdown Cooling Heat Exchangers (Relief Request #PSI-05)

Code Requirements

A surface and volumetric examination is required of pressure-retaining nozzle-to-vessel welds.

Code Relief Request

Relief was requested from performing 100% of the Code-required examination.

Reason for Request

A reinforcement collar has been welded to the shutdown heat exchanger shell and nozzle body making the pressure-retaining welds completely inaccessible. The applicant performed a surface examination of all reinforcement saddle-to-process pipe welds as an alternative examination. ASME Code, Section III, fabrication records are available on file.

Staff Evaluation

The existing design makes the nozzle-to-vessel welds of the shutdown heat exchanger inaccessible because the pressure-retaining weld required to be examined is totally covered by a reinforcement saddle.

The staff has determined that the fabrication examination performed on the pressure-retaining welds, the surface examination of the saddle attachment welds, and the hydrostatic test demonstrate an acceptable level of structural integrity.

(6) Reactor Coolant Pump Casing and Studs (Relief Request #PSI-06)

Code Requirement

Examination Category B-L-1 - A volumetric and surface examination is required for the pump casing welds.

Examination Category B-G-1 - A volumetric and surface examination is required for the pump studs.

Code Relief Request

Relief is requested to substitute shop fabrication examinations for the preservice inspection. During fabrication radiography and liquid penetrant examination were performed on pump casing welds. On the pump studs a magnetic particle examination was performed during fabrication and an ultrasonic examination was performed during the preservice inspection.

Reason for Request

Paragraph IWB-2200 of Section XI states in part:

> (b) Shop and field examinations may serve in lieu of the on-site preservice examinations provided:
>
> (1) in the case of vessels only, the examination is performed after the hydrostatic test required by Section III has been completed;
>
> (2) such examinations are conducted under conditions and with equipment and techniques equivalent to those that are expected to be employed for subsequent inservice examination;
>
> (3) the shop and field examination records are, or can be, documented and identified in a form consistent with those required in IWA-6000.

Staff Evaluation

The staff has determined that the fabrication and preservice examinations performed on the reactor coolant pump casing welds and studs were conducted under conditions and with equipment and techniques equivalent to those that are expected to be used for subsequent inservice examination. Therefore, the staff has concluded that the applicant has provided an acceptable alternative to the required preservice examinations.

(7) Visual Examination of Component Supports Based on Article IWF of Section XI (Relief Requests #PSI-07, PSI-08, PSI-09, PSI-10, and PSI-11)

Code Requirement

A visual examination is required using method VT-3 or VT-4.

Code Relief Request

Relief is requested to substitute fabrication examinations for the preservice inspection because the component supports are partially or completely inaccessible for examination.

Reason for Request

(a) Relief Request #PSI-07 - The component supports are in penetrations so that the supports are completely inaccessible for examination.

(b) Relief Request #PSI-08 - The component supports are partially blocked by adjacent U-bolts, which are not easily removed.

(c) Relief Request #PSI-09 - Component support access is partially blocked by fire-/heat-resistant insulation applied to protect supporting structural steel. The insulation is applied by spraying Flamastic or Pyrocrete over a wire mesh support. The insulation solidifies into a nonremovable mass approximately 3 in. to 5 in. thick. Fire-barrier integrity is a limiting condition for operation as identified in Technical Specification Paragraph 3.7.11. This creates undue hardship in conducting examinations. Accessible (uninsulated) areas of supports are examined.

(d) Relief Request #PSI-10 - Component supports are in penetrations that are closed off by permanently installed fire seals. Fire seal material is pumped into the penetration in a semiliquid state and solidifies into a nonremovable mass. Fire-seal integrity is a limiting condition for operation as identified in Technical Specification Paragraph 3.7.11. This creates undue hardship in conducting examinations.

(e) Relief Request #PSI-11 - Component support access is partially blocked by permanent (nonremovable) insulation. Supported lines operate at temperatures substantially below ambient and are, therefore, subject to severe condensation. The type of insulation used has a permanently sealed vapor barrier to exclude moisture, and removal of the insulation in the support area results in vapor contamination of the surrounding insulation. Possible alternate removable-type vapor barrier insulation is not acceptable for use because of the high fluoride/chloride content. The requirement for a vapor barrier seal necessitates nonremovable insulation. Accessible areas of supports are inspected.

Staff Evaluation

In the letter dated February 10, 1984, the applicant identified the specific welds or component supports that are partially or completely inaccessible for examination. On the basis of the review of this information, the staff has determined that the nondestructive examinations performed during construction exceed the visual inspections required by Section XI. Therefore, the staff has concluded that the applicant has provided an acceptable alternative to the Code requirement and, therefore, Relief Requests #PSI-07, PSI-08, PSI-09, PSI-10, and PSI-11 may be granted.

D.4 CONCLUSION

On the basis of the foregoing, the staff has determined, pursuant to 10 CFR 50.55(a)(2), that certain Section XI-required preservice examinations are impractical, and compliance with the requirements would result in hardships or unusual difficulties without a compensating increase in the level of quality and safety.

The technical evaluation has not identified any practical method by which Waterford 3 can meet all the specific preservice inspection requirements of Section XI of the ASME Code. To require exacting compliance with Section XI would delay the startup of the plant to redesign a significant number of plant systems, obtain sufficient replacement components, install the new components, and repeat the preservice examination of these components. Examples of components that would require redesign to meet the specific preservice examination provisions are the reactor vessel, shutdown cooling heat exchangers, and a significant number of the piping and component support systems. Even after the redesign effort, complete compliance with the preservice examination requirements probably could not be achieved. However, the as-built structural integrity of the existing primary pressure boundary has already been established by the construction code fabrication examinations.

On the basis of its review and evaluation, the staff concludes that public interest is not served by imposing certain provisions of Section XI of the ASME Code that have been determined to be impractical. Pursuant to 10 CFR 50.55(a)(2), the staff has allowed relief from these requirements which are impractical to implement and would result in hardship or unusual difficulties without a compensating increase in the level of quality and safety.

APPENDIX E

FEDERAL EMERGENCY MANAGEMENT AGENCY
INTERIM FINDINGS

Federal Emergency Management Agency

Washington, D.C. 20472

FEB 7 1984

MEMORANDUM FOR: Edward L. Jordan
Director
Division of Emergency Preparedness
and Engineering Response
Office of Inspection and Enforcement
U.S. Nuclear Regulatory Commission

FROM: Richard W. Krimm
Assistant Associate Director
Office of Natural and Technological
Hazards Programs

SUBJECT: Interim Finding on Waterford III Steam Electric Station

The Federal Emergency Management Agency (FEMA) transmits to the Nuclear Regulatory Commission (NRC) the attached Interim Finding on Waterford III Steam Electric Station dated September 16, 1983, an addendum to the Interim Finding dated December 27, 1983, and comments on the E.L. Quarantelli Report entitled: "Evacuation Behavior: Case Study of the Taft Louisiana Chemical Tank Explosion Incident."

These attachments include a response to the concerns raised by the St. John the Baptist Parish Civil Defense Director as requested in your memorandum of March 25, 1983.

FEMA Region VI staff and the State of Louisiana are continuing discussions on several unresolved elements. When a resolution to these issues has been reached, an addendum will be forwarded to your office. Based on the Region VI review of the Louisiana and St. John the Baptist and St. Charles Parishes' off-site radiological emergency preparedness plans, there is reasonable assurance that the plans are adequate and capable of being implemented in the event of an accident at the site. An exercise to test these plans is scheduled for February 8, 1984. A finding on preparedness will be made following this exercise.

Attachements
As Stated

Federal Emergency Management Agency

Region VI, Federal Center, 800 North Loop 288
Denton, Texas 76201-3698

January 17, 1984

MEMORANDUM FOR: RICHARD W. KRIMM, ASSISTANT ASSOCIATE DIRECTOR
Office of Natural and Technological Hazards

ATTENTION: Gloria Joyner, Program Specialist

FROM: R. Dell Greer, Chief
Natural and Technological Hazards Division

SUBJECT: Interim Findings for Waterford III
(Report of Professor E. L. Quarantelli entitled: "Evacuation Behavior: Case Study of the Taft Louisiana Chemical Tank Explosion Incident")

The attached review is to be included in previous submissions to complete the interim findings for Waterford III.

Region VI, at this time, sees no need to make any recommendations to Louisiana for plan changes around the Waterford III site due to the comments made in the Quarantelli report. Many of the problems sited in the report were covered by changes made to the plans since the Quarantelli report was made. Also problems will be eliminated due to the installation of the A/N system that has been completed since the report was made.

Region VI will be making a complete evaluation of the plans and the preparedness of the State and local parishes around Waterford III in the upcoming exercise to be held on February 8, 1984.

A complete exercise report on the Waterford III Exercise will be prepared and furnished to FEMA National as soon as possible after February 8, 1984.

Review of E. L. Quarantelli's final report of the Evacuation Behavior: Case Study of the Taft, Louisiana, Chemical Tank Expolsion Incident.

Throughout the report are discussions of the activities of the local emergency organizations, particularly their involvement in the large-scale evacuation that occurred as a result of the chemical explosion.

We have limited our response to Section VII of the report, "An Assessment of Actions in the Incident," since we feel this covers the major discussion items made throughout the report.

VII. An Assessment of Actions in the Incident

1. How well-prepared were the organizations and the community for the incident that occurred?

Discussion: The Quarantelli report states that for this locality, "There was better than average preparations." Therefore, we will not comment on this section except that FEMA will be evaluating the preparedness of the State and local parishes during the waterford exercise to be held on February 8, 1984, and will furnish a complete report of the exercise as soon as possible after its completion.

2. How well did the community and the organization learn about the threat?

Discussion: In the event that an accident happens at Waterford III, the public will be alerted by a siren system (now installed and operating, not officially tested) that covers the 10-mile EPZ. The sirens will be controlled and operated by parish emergency preparedness officials. Some fringe areas will be alerted by portable sirens and other means. A Public Information Brochure will be distributed to the public prior to the plant becoming operational. This brochure will describe to the public that if the siren system is sounded they are to listen to certain radio and T.V. stations for instructions on what actions they are to take. There are also direct communication link-ups between the utility, local and State emergency operating centers so that information on the conditions at the utility can be passed to the decisionmakers and then on to the public for actions to either evacuate the area, take shelter or other procedures.

3. How well was the evacuation organized?

Discussion: As previously mentioned, the Public Information Brochure will have a map showing evacuation routes that people living in certain sections are to follow to a known reception center. Also they are told to listen to Radio and T.V. stations for additional information on evacuation procedures to follow. This PIB was not in the hands of the public during this evacuation. In addition, prewritten notification messages and public information materials have been developed for the parish emergency plans. These messages specify the personal items that the public are to take with them, procedures to follow, and information about the reception centers to go to if told to evacuate. This information will be repeated regularly over the Emergency Broadcast System (EBS) radio and T.V. stations.

4. How well were evacuees sheltered?

Discussion: The plans developed for Waterford call for reception centers (already pre-selected and identified) to be located outside the 10-mile EPZ. These centers will be managed by emergency personnel of the parishes in which the centers are located. This should remove the only minor problem mentioned in the Quarantelli report that "the management of the shelters was criticized by some persons." The Quarantelli report had no major problems with this section of the evaluation; therefore, no further discussion will be offered on this.

5. How well handled was the return to normal?

Discussion: There are several points made in the Quarantelli report under this heading. One was the need for non-routine interaction among several key organizations and key decisionmakers at the plant. The emergency plans for Waterford already specify a precise network of communications between the State, local parishes, and the utility. The type of information to be passed and the responsible decisionmakers have been identified in advance, and technical support to the EOC is through established procedures.

Convergence at the local EOC's and dealing with the mass media personnel were additional problems.

In the future, security personnel will be stationed at the EOC's to allow entry to only those personnel who have proper identification. The waterford plans have an established method to cover the mass media situation; however, this procedure has not been tested as yet.

Federal Emergency Management Agency

Region VI, Federal Center, 800 North Loop 288
Denton, Texas 76201-3698

December 27, 1983

MEMORANDUM FOR: DAVE McLOUGHLIN, Deputy Associate Director
State and Local Programs and Support

ATTENTION: Gloria Joyner, Program Specialist
State and Local Programs and Support
Natural and Technological Hazards Division

FROM: for Jerry Stephens, Regional Director

SUBJECT: Addendum to Interim Findings on Waterford III Steam Electric Station

An interim finding on Waterford III Steam Electric Station was submitted to FEMA Headquarters on September 16, 1983. The plan review discovered that there were still remaining elements that proved to be inadequate or that needed further explanation. To resolve those remaining deficiencies, FEMA Region VI held a meeting November 8, 1983, in Dallas, Texas, with representatives from the State of Louisiana. Also in attendance were representatives from Louisiana Power and Light Company (LP&L), Argonne Lab, and Region VI RAC.

Attachment I provides a list of those unresolved elements that were specifically discussed at the November 8, 1983, meeting and progress made on resolving those elements. As noted, several of the elements have since been resolved while the remaining ones have been agreed upon but resolution not yet completed.

Attachment II is the formal submittal of the State of Louisiana comments to the Consolidated RAC Review (Interim Finding dated September 16, 1983) and also a response to concerns and resolutions pertaining to St. John Parish. FEMA Region VI is satisfied that all concerns pertaining to St. John Parish have been resolved.

You should note that the State of Louisiana included additional information and clarification on the following elements which were previously evaluated as adequate by FEMA Region VI. Those elements are as follows: A.1.d., C.2.a., D.4., F.1.d., G.1., G.4.a., H.10., I.8., J.10.i., J.10.1., J.12., K.4., O.1., P.3., P.8.

Also, please be advised that my staff is in the process of developing a written response pertaining to the Quarantelli Report per your memo dated November 23, 1983. Those comments will be forthcoming as soon as possible.

We will continue to maintain close liaison with the State of Louisiana to ensure that the remaining elements are completed to our satisfaction and will notify FEMA National accordingly.

Should you have any questions pertaining to this information, please contact Mr. Al Lookabaugh, Chief, Technological Hazards Branch.

Attachments

WATERFORD III

DISCUSSION OF UNRESOLVED ELEMENTS

A.2.a. Resolved

RAC comment: Most agencies do not mention key individuals by title.

Resolution: The State of Louisiana brought to our attention that key state individuals are specified in the State Implementing Procedures. Also key Parish individuals are specified in the Parish Implementing Procedures. A cross-reference to indicate this will be added to the State Plan.

A.3. Resolved

RAC comment: EPA is not listed among the organizations to support the plan.

Resolution: EPA was not listed because DOE and FEMA are specified as the lead agencies in the State plan. Support from other agencies will be coordinated through these two Federal agencies. Also, FEMA Region VI agrees that REACT is not expected to be used by the parishes in emergencies and references to REACT should be dropped in the next revision to the State Plan.

RAC comment: Letters of Agreement need to be formalized and updated before the Plan can be considered to be complete. This includes updating letters as needed.

Resolution: State of Louisiana forwarded to FEMA dated November 16, 1983, a copy of all Letters of Agreement that are currently on file at LNED. All letters will not be incorporated in the plans but rather a list will be used to illustrate which letters are on file. State of Louisiana agrees to update Letters of Agreement as necessary and verification by FEMA would be available for inspection. Relative to the ambulance service agreements for responding to an accident at Waterford III, an intra-parish mutual aid agreement currently exist which specifies general ambulance support between parishes. This agreement is through the Southeast Louisiana Emergency Medical System Council. FEMA Region VI has reviewed this mutual aid agreement document and approves of it.

C.1.b. In agreement but resolution not completed

RAC comment: Inadequate until plans/agreements are completed relating to specific Federal resources expected.

Resolution: The State of Louisiana has said that resource request will be specified when known to the Louisiana Nuclear Energy Division through final version of the Federal Radiological Monitoring and Assessment Plan (FRMAP). FEMA Region VI staff and RAC agree with the State of Louisiana.

C.1.c. In agreement but resolution not yet completed

RAC comment: Specific support resources are to be outlined in Letters of Agreement which have not been completed. Incorrect cross-references.

Resolution: FEMA Region VI staff and RAC agree with the State of Louisiana that only after the final version of the Federal Radiological Monitoring and Assessment Plan should specific State and local resources be available to support the Federal response. Reference to Letters of Agreement in Section VII.A.4., page 40, will be deleted in the next revision of the State Plan. Cross references will be corrected in the next plan revision.

C.3. In agreement but resolution not completed

RAC comment: Plan needs more detailed description pertaining to the capabilities and availability of the labs.

Resolution: An updated Letter of Agreement relative to the LSU Nuclear Science Department lab capability will be completed and amplified with the State of Louisiana. Also the concept of a mobile laboratory has been dropped by LNED and will be deleted in the next revision of the State Plan. Samples will be taken back to the Baton Rouge lab which is only an hour's drive.

C.4. Resolved:

RAC comment: No Letters of Agreement found in the Southern Mutual Radiological Assistance Plan. Also Letters of Agreement with hospitals need to be completed. No specific arrangements for emergency support by other local organizations or individuals could be found in plans.

Resolution: The Southern Mutual Radiological Assistance Plan constitutes an agreement (covered by law) that has been signed by the governors of the respective states. Letters of Agreement with the hospitals and nursing homes have been completed and will be submitted with the other letters. FEMA Region VI has since received the hospitals' Letters of Agreement. Request for outside resources is detailed in Parish Implementing Procedures and response time has been anticipated. Also State and Parish Implementing Procedures provide methods for detailing anticipated resource requirements at different emergency classifications. This information will be transmitted to the proper response organization prior to exhausting available resources. Thus, FEMA Region VI is satisfied that this element has been met.

E.1. Resolved

RAC comment: Message verification was not clear in the plans. Also EPA has no defined role in plan.

E.1. (Continued)

Resolution: The operational Hotline is a self-verifying notification system. Initiating calls can only be made from the plant. Also, each message form has a commercial telephone number available for verification. Also as stated in response to A.e., DOE and FEMA are the lead Federal agencies. Any supporting agencies will be notified through these two Federal agencies. Thus, FEMA Region VI is satisfied that this element has been met.

E.5.

Resolved

RAC comment: It is not clear that a joint public information center coordinates the information to be released. Also it should be made clear which public information officers can approve information for release at local level.

Resolution: Federal guidance does not require a joint public information center. Protection action messages will be released by local and State organizations via local media and EBS as appropriate. The St. Charles and St. John emergency plans call for the release of emergency public information through their respective parish public information officers. It is specified in the Parish Implementing Procedures that only the Parish President can authorize public information releases. Thus, FEMA Region VI is satisfied that this element has been met.

H.11.

In agreement but resolution not completed

RAC comment: What is the concept on kits? Plan might benefit by describing what portion of this equipment is in kits and where those kits are.

Resolution: State of Louisiana states that emergency kits are in a foot locker. Some items are used regularly and are not locked in a kit. There will be a change in the plans to include a listing of all items. Will also change wording in the plan from "sampling supplies" to "LNED Emergency Response Kits."

I.10.

In agreement but resolution not completed

RAC comment: Alternative methods for estimating dose should be described in the plan. Also the computer may not be available when needed.

Resolution: The procedures for estimating dose are those incorporated by EPA-520/1-75-001, Appendix D. A hand method for estimating doses will be included in the next revision of the State Implementing Procedures.

J.2.

Resolved

RAC comment: State Plan does not provide for provisions concerning on-site individuals at the plant.

J.2. (Continued)

Resolution: This criteria refers to the evacuation of on-site personnel to suitable off-site locations. It does not refer to arrangements for reception or sheltering of the general public in support parishes. Information is provided in Chapter 4.VI.F., enclosures 1 and 2, demonstrates coordination between the Waterford III Plan and local plans for movement and handling of on-site personnel who may need to be evacuated to an off-site location. Appropriate cross references to State Plan should be added to indicate this information is located in the parish enclosures. Thus, FEMA Region VI is satisfied that this element has been met.

J.9.

In agreement but resolution not completed

RAC comment: Section IV.A.6.b. of Chapter 7 needs to be revised. The dose levels mentioned there can in no way be considered "limits for routine operations" as stated. Also the note on page 8-5 regarding the bases for the PAG's needs to be expanded or placed elsewhere in the text.

Resolution: State of Louisiana explained the dose levels considered "limits for routine operations" and the EPA RAC representative then agreed. In next plan revision (Chapter 7, IV.A.6.b., page 7-7, the term "for routine operations" will be changed to "for the general population." Also (Chapter 7, IV.B.2.b.(1) page 7-9) the term "available" will be changed to "warranted." A correction was agreed upon to change the note on page 8-5 to indicate that such note is not correct for FDA which refers to critical receptor but is correct for contaminated drinking water supplies. Not correct for food preventive PAG's. Appropriate changes will be made in the next plan revision by the State of Louisiana.

J.10.e.

In agreement but resolution not completed

RAC comment: Nowhere in either the State or parish plans does it provide for the quantities and storage of KI. Also, additional cross-references needed.

Resolution: Next revision of plan will include a statement "Quantities of KI, sufficient to meet short term off-site contingencies, will be made available to St. Charles and St. John Parishes by Louisiana Power and Light for storage in their EOC's, and will be administered at the order of the ASOEA in accordance with State policy.

J.10.m.

In agreement but resolution not completed

RAC comment: Interpretation of projected dose must be clearly understood by the decisionmakers and carefully spelled out in the plan.

J.10.m. (Continued)

Resolution: State of Louisiana has agreed to put a full definition of projected dose in a footnote, referenced, and defined in Tab 1, Chapter 6 and 7, in the next revision of the State Plan.

L.1.

In agreement but resolution not complete

RAC comment: There should be a statement in the plan to verify the capability of Ochsner Clinic.

Resolution: State of Louisiana agreed to put a statement of capability in the next revision of the State plan.

RAC comment: Who is responsible for training?

Resolution: The entire issue surrounding training and who is responsible for specific training is still unresolved The Southeast Louisiana Emergency Medical Systems Council is very interested in providing training along with LNED. A meeting is to be held the week of December 26 to determine who will be responsible for conducting specific training.

RAC comment: Are agreements signed with local ambulance services for responding to an accident at Waterford III? None were in the plans.

Resolution: Intra-parish mutual aid agreements have been completed concerning ambulance support between parishes. FEMA Region VI now has a copy of the ambulance agreement and approves it as being acceptable.

ATTACHMENT II

November 16, 1983

Mr. Al Lookabaugh
FEMA, Region VI
800 N. Loop 288
Denton, Texas 76201-3698

Dear Mr. Lookabaugh:

Subsequent to the meeting held on November 8, 1983, enclosed is the formal submittal of the State of Louisiana comments to the Consolidated RAC Review of the Louisiana Peacetime Radiological Response Plan, Revision 4, and Attachment 1. A few of the items discussed at the meeting remain open or are awaiting completion. Please find enclosed, in **bold print,** the items identified during the meeting which require changes to the State Plan or Attachment 1.

Also enclosed, is our response to your Attachment 1 of the Consolidated RAC Review dated September 28, 1983, St. John the Baptist Parish Concerns and Resolutions.

If there are any questions or further information needed, please contact Mr. Thomas Laiche at the address shown below.

Sincerely,

William H. Spell

WHS:TL:st

Enclosures

RESPONSE TO RAC REVIEW COMMENTS OF SEPTEMBER 28, 1983

A.1.d The Director of the Bureau of Emergency Medical Services is identified in the Department of Health and Human Resources, Office of Hospitals, Bureau of EMS Implementing Procedures.

A.2.a Key state individuals are specified in the state Implementing Procedures (IP's).

Key parish individuals are specified in the parish Implementing Procedures (IP's). **A cross reference will be added to the State Plan to indicate this.**

A.3 DOE and FEMA are specified as the lead agencies in the state plan. Support from other agencies will be coordinated through these agencies.

A list of Letters of Agreement will be added to the Plan. Copies of the letters and any verifying statements will be made available upon request.

REACT is not expected to be used by the parishes in emergencies and references to REACT will be dropped in the next revision to the State Plan.

C.1.b **Resources will be specified, when made known to the Louisiana Nuclear Energy Division through final version of the Federal Radiological Monitoring and Assessment Plan (FRMAP).**

C.1.c **State and local resources available to support the Federal response, will be outlined when Federal response resources and anticipated support needed are specified through final version of FRMAP.**

Reference to letters of agreement in Section VII.A.4, page 40 will be deleted in the next revision of the State Plan.

Correct cross reference as specified.
Attachment, page iii.
Change page number.

C.2.a **Correct cross reference as specified.**
Attachment, page iii.
Change page number.

C.3 **State Plan**
Tab 3 to Chapter 6
G.2. page 6-13
Delete sentence which describes mobile laboratory.

Add a description of the LSU Nuclear Science Department capability to support LNED's emergency response.

11-23-83

C.4 Southern Mutual Radiological Assistance Plan (SMRAP) constitutes an agreement (see Chapter 2 of SMRAP) and has been signed by the governors of the respective states.

LOA with Hospitals and Nursing Homes have been completed and will be submitted with the other letters.

Request for outside resources is detailed in parish IP's and response time has been anticipated.

State and parish IP's provide methods for detailing anticipated resource requirements at different emergency classifications. This information will be transmitted to the proper response organization prior to exhausting available resources.

D.4 **Add a cross reference to the State Plan that indicates this information this information is also available in State IP's.**

E.1 The Operational Hotline is a self-verifying notification system. Initiating calls can only be made from the plant. Also, each message form has a commercial telephone number available for verification.

As stated in response A.3, DOE and FEMA are the lead federal agencies. Any supporting agencies will be notified through these. Federal resource requirements will be listed as soon as they are made available to the LNED.

E.5 Federal guidance does not require a joint public information center. Protective action messages will be released by local and state organizations via local media and EBS as appropriate. The St. Charles and St. John emergency plans call for the release of emergency public information through their respective Parish Public Information Offices. It is specified in the parish IP's that only the Parish President can authorize public information releases.

F.1.d **Correct cross reference as specified.**

State Plan, page vii	**Attachment, page iv**
Add page number 3-3	**Enclosure 2, change letter I to H**

G.1 **Correct cross reference**
Attachment, page iv
G.1., add page number 24

G.4.a St. John the Baptist and St. Charles parishes reserve the right to maintain independent public information organizations. Information released is specific to the individual parishes. A TWX capability has been established specifically for coordination of public information between organizations.

The Parish President, as the chief elected official, reserves the right by home rule charter to make this decision. There may be situations where the designated spokesperson is not the public information officer.

H.10 Correct cross reference as specified.
State Plan, page vii
Add - H.10 Chapter 6, page 6-13, Tab 3,G

H.11 Enclosure 1 to Tab 3 of Chapter 6
11.B. page 6-17
Change title Sampling Supplies to LNED Emergency Response Kits

These kits are maintained and inventoried in the LNED laboratory after use or semi-annually.

Parish emergency equipment is supplied and maintained by LOEP and is inventoried at each parish EOC after use or semi-annually.

I.8 **Add anticipated response times for LNED personnel**
Add a cross reference to State Plan to show that call our list for LNED personnel is located in the State implementing procedures

I.10 The procedures used are those incorporated by EPA-520/1-75-001, Appendix D.

A hand method for estimating off-site dose projections will be added to State implementing procedures.
Add a cross reference to the State plan that indicates this information is available in the State IP's.

J.2 This criteria refers to the evacuation of onsite personnel to suitable offsite locations. It does not refer to arrangements for reception or sheltering of the general public in support parishes. The information provided in Chapter 4.VI.F, enclosures 1 and 2, demonstrates coordination between the W3 Site Plan and local plans for movement and handling of onsite personnel who may need to be evacuated to an offsite location.

Add a cross reference to the State plan to indicate this information is located in the Parish Enclosures

J.9 The statement is intended to say that limitations to exposure for emergency workers will be imposed when radiation doses approach the 5 rem threshold. The intention is to be more conservative, rather than allow emergency workers doses to reach 25 rem.

Chapter 7, IV.A.6.b., page 7-7, change the term "for routine operations" to "for the general population."

Chapter 7, IV.B.2.b.(1) page 7-9, change the term "available" to "warranted".

J.10.e Correct cross reference as specified.
State Plan, page viii
Add - Chapter 9, V.B.2, page 9-9
Table to Chapter 9, page 9-13

Change the following:
Chapter 5 to Attachment I V.B.2.b., page 46
Delete the second sentence which reads, "This substance will be supplied by LNED..." Add the following: "Quantities of KI, sufficient to meet short term offsite contingencies, is available at St. Charles Parish and St. John the Baptist Parish EOC's, and will be administered at the order of the ASOEA in accordance with state policy

J.10.i and J.10.1. The W3, Evacuation Time Estimate is referenced in the emergency plans for the respective parishes and is available to those decision makers who will locate in the Parish EOc's.

J.10.m Tabs 1 and 2, Chapter 6, pages 6-7 through 6-10 explain the concept of PAG's. However, the PAG's are not the only criteria used in determining protective actions. The rish parishes use considerable flexibility in making decisions for protective actions.

A full definition of projected dose as stated in EPA-520/1-75-0001, September 1975, page 2.1 - 2.2 will be included in Tab 1, Chapter 6 and Tab 1, Chapter 7 of the State Plan.

J.12 Arrangement for the registering and monitoring of evacuees are available in the support parish plans. The radiation monitoring equipment is also described in support parish plans. Equipment is stored in the support parish Civil Defense offices, with back-up units available through the Louisiana Office of Emergency Preparedness.

K.4 State Plan Chapter 9, III.E. page 9-3, lines 4 and 5:
Change the work "will" to "may".

L.1

1. A statement to verify Ochsner's capability will be included in the revision of the State Plan.

2). Training for local and back-up medical services is provided for by the Southeast Louisiana Emergency Medical Systems Council.

3.) Intra parish mutual aid agrement exist which specifies general ambulance support between parishes. Training will be provided by the Southeast Louisiana Emergency Medical Systems Council.

4.) At this time, the State is re-evaluating its' position with regards to the use of the local hospitals to handle contaminated individuals. Major hospitals that are near the Nuclear facilities are more capable of handling contamination problems. Training at the major hospitals can be more comprehensive than trying to train a large number of smaller, local hospitals that may not be able to cope with a contamination situation. When a more definite decision is made by the state, you will be notified. Training will be provided for through the state and the Southeast Louisiana Emergency Medical Systems Council.

5). See answer number 4 above.

6). See answer number 3 above.

7). St. Charles and St. John the Baptist parishes are unique in their need and development of emergency plans. Yes, the EMS system was involved in the planning stages.

8). NUREG 0654 section L.1. requires the hospital and medical support be arranged for, and that personnel are trained for this support role. It is our opinion that a description of how a local plan interfaced with the EMS system and how the parishes arrived at their needs for medical manpower is not required for inclusion in the plans.

9). Medical attendants are provided with ambulances as a normal business procedures. Again, training for drivers and attendants is provided for by the Southeast Louisiana Emergency Medical Systems Council in coordination with LNED.

10). See answer number 1 above.

O.1 LNED has the responsibility of training. At this time, LNED and the licensee are developing a training program and timetable for upcoming training.

P.3 **Correct cross reference as specified.**
State Plan, page ix
Change page number from 22 to 26

P.8 **Correct cross reference as specified.**
Attachment
Add page numbers iv through viii

ST. JOHN THE BAPTIST PARISH CONCERNS AND RESOLUTIONS

1. Frequent malfunction of the operational hotline phone.

The initial problems encountered with the operational hotline have been resolved. The proper operation of the hotline is being confirmed through monthly tests leading to the Waterford 3 exercise-for-score. Following the exercise, the operational hotline will be tested in accordance with the guidance established in NUREG-0654. Any malfunctions discovered as part of the testing program will promptly be remedied by LP&L.

In addition, a push-to-talk feature and a mouthpiece confidencer device have been installed at St. John's hotline station to reduce background noise from being transmitted through the system. Also, a feature is to be installed which will allow each hotline station to ring-up the Waterford site during an emergency.

2. Prompt notification of individuals in the fish camps within the 10-mile EPZ.

LP&L has purchased a portable siren for St. John Parish which will be capable of notifying 75% of the camps located in the wetlands. LP&L is in the process of purchasing two helicopter mounted warning devices for St. John Parish and two for St. Charles Parish.

The Louisiana Nuclear Energy Division has made contact with three State agencies who operate helicopters: the Louisiana State Police, the Louisiana Department of Wildlife and Fisheries, and the Louisiana Department of Transportation and Development. Each of these agencies has given assurance that helicopters will be made available in the event of an emergency. In addition, St. John Civil Defense is seeking an agreement from a private provider for two helicopters to be used in an emergency. These private helicopters are located several miles beyond the perimeter of the 10 mile EPZ and could be made available on short notice.

Federal Emergency Management Agency

Region VI Federal Center Denton, Texas 76201

September 16, 1983

MEMORANDUM FOR: DAVE MC LOUGHLIN
Acting Associate Director
State and Local Programs and Support

FROM: Jerry Stephens
Regional Director

SUBJECT: Interim Finding on Waterford III Steam Electric Station

Attached is a copy of the Federal Emergency Management Agency Region VI Radiological Assistance Committee, Argonne National Laboratory, and FEMA Region VI review of the State of Louisiana Peacetime Radiological Response Plan Revision #4 and the St. Charles and St. John the Baptist Parishes' emergency response plans. These off-site plans were developed and submitted to FEMA Region VI in accordance with Paragraph 350.7 of 44 CFR, Part 350 in support of the Waterford Plant.

The review of the plans was based on Section II (A through P), Planning Standards and Evaluation Criteria, NUREG-0654/FEMA-REP-1, Rev. 1.

Also in response to a memorandum dated March 25, 1983, from Edward L. Jordan to Richard W. Krimm, FEMA was requested to review the five concerns expressed by the St. John Parish Civil Defense Director and include our findings as a part of this interim finding.

We also had a concern brought up by Mr. Charles Hackney (NRC Regional Office, Arlington, Texas) to my RAC Chairman concerning how the personnel on the ships that are docked along the Mississippi (loading or unloading cargo) would be evacuated.

This item was discussed by the RAC Chairman with State and local personnel who advised that the ships' personnel would be considered as part of the industry where the ships were docked. Therefore, the ships' personnel would be evacuated using the evacuation plan for that particular industry.

The inadequate elements discovered by the review of the State and Local Plans will be furnished to the State of Louisiana by letter for comment and/or corrections. We will maintain close liaison with the State to see that the inadequate elements are corrected to our satisfaction and will notify FEMA National at that time.

Based on the review of the State and Parish Off-site Emergency Response Plans, there is reasonable assurance that the plans are adequate and capable of being implemented.

Many of the remarks in the review of the plans indicate that several elements are inadequate due to the lack of letters of agreement. The State has assured FEMA that most of these letters have already been obtained and they are in the process of obtaining the remainder. They wished to obtain all letters before submitting them to FEMA.

Attachments

APPENDIX F

PRINCIPAL CONTRIBUTORS TO SSER NO. 6

Name	Branch
R. Anand	Auxiliary Systems
H. Balukjian	Core Performance
L. Bender	Licensee Qualification
R. Benedict	Licensee Qualification
K. Campe	Siting Analysis
F. Clemenson	Auxiliary Systems
J. Gilray	Quality Assurance
J. Hayes	Meteorology and Effluent Treatment
G. Hsu	Core Performance
T. Huang	Core Performance
M. Hum	Materials Engineering
J. Jackson	Equipment Qualification
C. Liang	Reactor Systems
W. Long	Procedures and Systems Review
G. McPeek	Standardization and Special Projects
D. Perrotti	Emergency Preparedness Licensing
F. Rinaldi	Structural and Geotechnical Engineering
R. Stevens	Instrumentation and Control Systems
D. Terao	Materials Engineering
J. Wermeil	Auxiliary Systems
M. Wigdor	Instrumentation and Control Systems
C. Willis	Meteorology and Effluent Treatment
F. Witt	Chemical Engineering
J. Wright	Equipment Qualification
S. Wu	Core Performance

APPENDIX I

CONTROL OF HEAVY LOADS AT NUCLEAR POWER PLANTS - WATERFORD GENERATING STATION, UNIT 3 - PHASE I

EGG-HS-6291
Revision 1

CONTROL OF HEAVY LOADS AT NUCLEAR POWER PLANTS
WATERFORD GENERATING STATION, UNIT 3
LOUISIANA POWER AND LIGHTING COMPANY
(Phase I)
Docket No. [50-382]

Author
J. P. Sekot

Principal Technical Investigator
T. H. Stickley

Published
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EG&G Idaho, Inc.
Idaho Falls, Idaho 83415

Prepared for the
U.S. Nuclear Regulatory Commission
Under DOE Contract No. DE-AC07-76ID01570

FIN No. A6457

ABSTRACT

The Nuclear Regulatory Commission (NRC) has requested that all nuclear plants, either operating or under construction, submit a response of compliancy with NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." EG&G Idaho, Inc., has contracted with the NRC to evaluate the responses of those plants presently under construction. This report contains EG&G's evaluation and recommendations for the Louisiana Power and Light Company (LP&L) Waterford Generating Station, Unit 3 (WGS No. 3).

EXECUTIVE SUMMARY

WGS No. 3 has demonstrated consistency with the intent of the guidelines of NUREG-0612.

CONTENTS

TABLES

CONTROL OF HEAVY LOADS AT NUCLEAR POWER PLANTS

WATERFORD GENERATING STATION, UNIT 3

(Phase I)

1. INTRODUCTION

1.1 Purpose of Review

This technical evaluation report documents the EG&G Idaho, Inc., review of general load-handling policy and procedures at Waterford Generating Station (WGS) No. 3. This evaluation was performed with the objective of assessing conformance to the general load-handling guidelines of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants" [1], Section 5.1.1.

1.2 Generic Background

Generic Technical Activity Task A-36 was established by the U.S. Nuclear Regulatory Commission (NRC) staff to systematically examine staff licensing criteria and the adequacy of measures in effect at operating nuclear power plants to assure the safe handling of heavy loads and to recommend necessary changes to these measures. This activity was initiated by a letter issued by the NRC staff on May 17, 1978 [2], to all power reactor applicants, requesting information concerning the control of heavy loads near spent fuel.

The results of Task A-36 were reported in NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." The staff's conclusion from this evaluation was that existing measures to control the handling of heavy loads at operating plants, although providing protection from certain potential problems, do not adequately cover the major causes of load-handling accidents and should be upgraded.

In order to upgrade measures for the control of heavy loads, the staff developed a series of guidelines designed to achieve a two-phase objective using an accepted approach or protection philosophy. The first portion of the objective, achieved through a set of general guidelines identified in NUREG-0612, Article 5.1.1, is to ensure that all load-handling systems at nuclear power plants are designed and operated such that their probability of failure is uniformly small and appropriate for the critical tasks in which they are employed. The second portion of the staff's objective, achieved through guidelines identified in NUREG-0612, Articles 5.1.2 through 5.1.5, is to ensure that, for load-handling systems in areas where their failure might result in significant consequences, either (a) features are provided, in addition to those required for all load-handling systems, to ensure that the potential for a load drop is extremely small (e.g., a single-failure-proof crane) or (b) conservative evaluations of load-handling accidents indicate that the potential consequences of any load drop are acceptably small. Acceptability of accident consequences is quantified in NUREG-0612 into four accident analysis evaluation criteria.

The approach used to develop the staff guidelines for minimizing the potential for a load drop was based on defense in depth and is summarized as follows:

- Provide sufficient operator training, handling system design, load-handling instructions, and equipment inspection to assure reliable operation of the handling system

- Define safe load travel paths through procedures and operator training so that, to the extent practical, heavy loads are not carried over or near irradiated fuel or safe shutdown equipment

- Provide mechanical stops or electrical interlocks to prevent movement of heavy loads over irradiated fuel or in proximity to equipment associated with redundant shutdown paths.

Staff guidelines resulting from the foregoing are tabulated in Section 5 of NUREG-0612.

1.3 Plant-Specific Background

On December 22, 1980, the NRC issued a letter [3] to Louisiana Power and Lighting Company (LP&L) the applicant for WGS No. 3 requesting that the applicant review provisions for handling and control of heavy loads at WGS No. 3, evaluate these provisions with respect to the guidelines of NUREG-0612, and provide certain additional information to be used for an independent determination of conformance to these guidelines. On June 19, 1981, LP&L provided the initial response [4] to this request.

On September 21, 1981, LP&L submitted a second or follow-up response to this request. Only Phase I guidelines will be addressed in this report. These involve approximately 60% of the June 19, 1981, response. The remaining sections of the June 19, 1981, and all of the September 21, 1981, response are concerned with Phase II. Compliance to Phase II requirements are semi-independent on Phase I and will not be addressed in this report. Based on the information submitted, a preliminary draft of this report was prepared and discussed with the applicant. Additional information [5] was provided on January 27, 1983. The final report (May 1983) was prepared from information contained in those submittals. This report identified inconsistencies with regard to guideline 4 (Special Lift Devices). Additional information [10] was provided on February 9, 1984. This revision to the final report is based on this additional information.

2. EVALUATION AND RECOMMENDATIONS

2.1 Overview

The following sections summarize LP&L's review of heavy load handling at WGS No. 3 accompanied by EG&G's evaluation, conclusions, and recommendations to the applicant for bringing the facilities more completely into compliance with the intent of NUREG-0612. The applicant has indicated the weight of a heavy load for this facility (as defined in NUREG-0612, Article 1.2) as 1500 pounds [11].

2.2 Heavy Load Overhead Handling Systems

This section reviews the applicant's list of overhead handling systems which are subject to the criteria of NUREG-0612 and a review of the justification for excluding overhead handling systems from the above-mentioned list.

2.2.1 Scope

"Report the results of your review of plant arrangements to identify all overhead handling systems from which a load drop may result in damage to any system required for plant shutdown or decay heat removal (taking no credit for any interlocks, technical specifications, operating procedures, or detailed structural analysis) and justify the exclusion of any overhead handling system from your list by verifying that there is sufficient physical separation from any load-impact point and any safety-related component to permit a determination by inspection that no heavy load drop can result in damage to any system or component required for plant shutdown or decay heat removal."

A. Summary of Applicant's Statements

The applicant's review of overhead handling systems identified the cranes and hoists shown in Table 2.1 as those which handle heavy loads in the vicinity of irradiated fuel or safe shutdown equipment.

TABLE 2.1 OVERHEAD HANDLING DEVICES IN VICINITY OF SAFE SHUTDOWN EQUIPMENT, WATERFORD GENERATION STATION UNIT 3

Handling System	Capacity (Tons)	Location
Reactor Circular Bridge	200/30	Reactor Building
Fuel-Handling Building Bridge	125/15	Fuel-Handling Building

The applicant has also identified numerous other cranes that have been excluded from satisfying the criteria of the general guidelines of NUREG-0612. These are listed in Table 2.2. These overhead handling devices were reviewed by the applicant to the criteria of NUREG-0612 and were excluded based on sufficient physical separation from any load impact point that could damage any system or component required for plant shutdown or decay heat removal. Some of the devices have been excluded because the applicant has indicated that the heavy load of approximately 1450 pounds for this facility would not be exceeded. Tables 2.3 and 2.4 identify heavy loads to be handled by each crane, load weight, designated lift device, procedure, and load-drop analysis.

B. EG&G Evaluation

The applicant's response [5] indicates that each overhead handling device at WGS No. 3 is listed in Tables 2.1 and 2.2. The applicant provided a listing of all plant overhead handling systems, identified equipment to be handled, crane or hoist location, elevations, and rated capacities. Drawings were also provided to show the proximity of the handling devices to safe shutdown equipment. The applicant addressed each handling system and provided justification for its exclusion from the list of OHS from which load drops may result in damage to any system required for plant shutdown or decay heat removal. They further addressed the handling of heavy loads identified in NUREG-0612 (Table 3.1-1).

TABLE 2.2 OVERHEAD HANDLING SERVICES EXCLUDED FROM FURTHER CONCERN, WATERFORD GENERATING STATION UNIT 3

Handling System	Capacity (Tons)	Location
Cranes		
Radwaste Cask-Handling Bridge	30	Reactor Auxiliary Building
Machine Shop Bridge	6	Service Building
Steam Generator Feeder Pump Bridge	10	Turbine Building
Intake Structure Bridge	40	Intake Structure
Turbine Building Gantry	200/35	Turbine Building
Monorail/Hoist		
Roof Hatch Cover	10	Reactor Auxiliary Building
Water Chiller	7-1/2	Reactor Auxiliary Building
Water Chiller (2)	7-1/2	Reactor Auxiliary Building
HVAC Fan Motors	7-1/2	Reactor Auxiliary Building
Cask Handling	7-1/2	Reactor Auxiliary Building
CEA Drive-MG Set	7-1/2	Reactor Auxiliary Building
RSD Equipment Access	7-1/2	Reactor Auxiliary Building
Emergency Diesel Generator (4)	3	Reactor Auxiliary Building
Emergency Diesel Generator (4)	14	Reactor Auxiliary Building
Purification Filter	5	Reactor Auxiliary Building
Misc. Equipment Jib Crane	1/2	Plant Shack
Spent-Fuel Handling Machine	3/4	Fuel-Handling Building
Refueling Machine	3/4	Reactor Building
Fuel-Pool Filter	5	Reactor Auxiliary Building

TABLE 2.2 (continued)

Handling System	Capacity (Tons)	Location
Monorail/Hoist (continued)		
Boric Acid Precon Filter	5	Reactor Auxiliary Building
Waste, Oil, & Laundry Filter	5	Reactor Auxiliary Building
Charging Pumps	5	Reactor Auxiliary Building
HP-LP Safety Injection Cont. Spray Pumps (2)	5	Reactor Auxiliary Building
HP Safety Inject, Drain Pump	5	Reactor Auxiliary Building
Safety VA Maintenance (2)	5	Reactor Auxiliary Building
Equipment Decon Room	5	Reactor Auxiliary Building
Equipment Decon Room (4)	1	Reactor Auxiliary Building
Equipment Hot Machine Shop	2	Reactor Auxiliary Building
General Storage (Above Machine Shop)	1	Reactor Auxiliary Building
Miscellaneous Equipment	1	Reactor Building
1PH Drain Pump (3)	5	Turbine Building
Chillers (2)	5	Chiller Building

TABLE 2.3 REACTOR CONTAINMENT BUILDING POLAR CRANE--WATERFORD GENERATING STATION UNIT 3

	Load	Approximate Weight (Ton)	Lift Equipment	Procedure	Remarks
1.	Reactor Vessel Head w/Lift Rig	189	Reactor Vessel Head Lift Rig	a	Load drop analysis over Reactor Vessel[b]
2.	Reactor Internals Lifting Rig	16.5	N/A	a	Less critical than (1)
3.	Reactor Upper Guide Structure w/Lift Rig	73	Upper Guide Structure Lift Rig	a	Less critical than (4)
4.	Reactor Core Barrel w/Lift Rig	79	Core Support Barrel Lift Rig	a	Load drop analysis[c] over canal bottom
5.	Stud Tensioner	1.5	e	a	Less critical than (4)
6.	RC Pump 1A-Motor w/Lift Rig	59	Reactor Coolant Pump Motor Lift Rig	a	Load drop analysis[d] over operating floor
7.	RC Pump 1B-Motor w/Lift Rig	59	Reactor Coolant Pump Motor Lift Rig	a	Load drop analysis[d] over operating floor
8.	RC Pump 2A-Motor w/Lift Rig	59	Reactor Coolant Pump Motor Lift Rig	a	Load drop analysis[d] over operating floor
9.	RC Pump 2B-Motor w/Lift Rig	59	Reactor Coolant Pump Motor Lift Rig	a	Load drop analysis[d] over operating floor
10.	Plant Equipment from Lower Floors	5	e	a	Load drop analysis[d] over operating floor
11.	Main Hook Load Block	4.5	N/A	a	Less critical than (10)
12.	Auxiliary Hook Load Block	1	N/A	a	Less critical than (10)

a. Procedures will be developed and implemented to cover load-handling operations for heavy loads that are or could be handled over or in proximity to irradiated fuel or safe shutdown equipment. These procedures will include the information identified in NUREG-0612 Section 5.1.1.(2). [5]

b. Analysis currently being performed. Results will be reported in Applicant's Report Part II as Appendix B. [5]

c. Analysis has been performed. Results will be reported in Applicant's Report Part II as Appendix D. [5]

d. Analysis has been performed and consequential effects have been found acceptable by the applicant. [5]

e. No special lifting devices identified by the applicant.

f. The spent-fuel cask cannot be brought over the spent-fuel storage pool; also, it cannot be lifted more than 30 feet from the floor. Both the spent-fuel cask storage and wash-down areas are supported on the huge mass concrete slabs which are structurally independent of the spent-fuel storage pool. Therefore, the impact due to a cask drop on the storage- and wash-down-area slabs will not have a detrimental structural effect on the spent-fuel storage pool structures. No other load drop analysis is required. [4]

TABLE 2.4 FUEL-HANDLING BUILDING BRIDGE CRANE--WATERFORD GENERATING STATION UNIT 3

Load	Approximate Weight (Ton)	Lift Equipment	Procedure	Remarks
1. Spent-Fuel Cask w/10 Fuel Assemblies	100	e	a	f
2. Gate No. 1	1.6	e	a	Less critical than (4)
3. Gate No. 2	1.6	e	a	Less critical than (4)
4. Gates No. 3A and 3B	12.7	e	a	Load drop analysis over storage area bottom
5. Gate No.	10.8	e	a	Less critical than (4)
6. Hatch Cover HC-6	11.5	e	a	Load will be handled at minimum height from floor
7. Hatch Cover HC-5	12	e	a	Load will be handled at minimum height from floor
8. Hatch Cover HC-15	5.5	e	a	Load will be handled at minimum height from floor
9. New Fuel Containers w/2 Fuel Assemblies	3.5	e	a	Load will be handled at minimum height from floor
10. Plant Equipment from Lower Floor	10.5	e	a	Load will be handled at minimum height from floor
11. Main Hook Load Block	2.1	e	a	Load drop analysis over operating floor

a. Procedures will be developed and implemented to cover load-handling operations for heavy loads that are or could be handled over or in proximity to irradiated fuel or safe shutdown equipment. These procedures will include the information identified in NUREG-0612 Section 5.1.1.(2). [5]

b. Analysis currently being performed. Results will be reported in Applicant's Report Part II as Appendix B. [5]

c. Analysis has been performed. Results will be reported in Applicant's Report Part II as Appendix D. [5]

d. Analysis has been performed and consequential effects have been found acceptable by the applicant. [5]

e. No special lifting devices identified by the applicant.

f. The spent-fuel cask cannot be brought over the spent-fuel storage pool; also, it cannot be lifted more than 30 feet from the floor. Both the spent-fuel cask storage and wash-down areas are supported on the huge mass concrete slabs which are structurally independent of the spent-fuel storage pool. Therefore, the impact due to a cask drop on the storage- and wash-down-area slabs will not have a detrimental structural effect on the spent-fuel storage pool structures. No other load drop analysis is required. [4]

C. EG&G Conclusions and Recommendations

Based on the information provided, EG&G concludes that the applicant has included all applicable hoists and cranes in their list of handling systems in compliance with the requirements of the general guidelines of NUREG-0612.

2.3 General Guidelines

This section addresses the extent to which the applicable handling systems comply with the general guidelines of NUREG-0612, Article 5.1.1. EG&G's conclusions and recommendations are provided in summaries for each guideline.

The NRC has established seven general guidelines which must be met in order to provide the defense-in-depth approach for the handling of heavy loads. These guidelines consist of the following criteria from Section 5.1.1 of NUREG-0612:

- Guideline 1--Safe Load Paths
- Guideline 2--Load-Handling Procedures
- Guideline 3--Crane Operator Training
- Guideline 4--Special Lifting Devices
- Guideline 5--Lifting Devices (not specially designed)
- Guideline 6--Cranes (Inspection, Testing, and Maintenance)
- Guideline 7--Crane Design.

These seven guidelines should be satisfied for all overhead handling systems and programs in order to handle heavy loads in the vicinity of the reactor vessel, near spent fuel in the spent-fuel pool, or in other areas where a load drop may damage safe shutdown systems. The succeeding paragraphs address the guidelines individually.

2.3.1 Safe Load Paths [Guideline 1, NUREG-0612, Article 5.1.1(1)]

"Safe load paths should be defined for the movement of heavy loads to minimize the potential for heavy loads, if dropped, to impact irradiated fuel in the reactor vessel and in the spent-fuel pool, or to impact safe shutdown equipment. The path should follow, to the extent practical, structural floor members, beams, etc., such that if the load is dropped, the structure is more likely to withstand the impact. These load paths should be defined in procedures, shown on equipment layout drawings, and clearly marked on the floor in the area where the load is to be handled. Deviations from defined load paths should require written alternative procedures approved by the plant safety review committee."

A. Summary of Applicant's Statements

The applicant submitted drawings identifying safe load paths, location of spent fuel, and safety-related equipment. Crane travel over areas not defined as safe load paths (i.e., exclusion areas) is prohibited without safety review. Safe load paths and exclusion areas will be defined in all load-handling procedures and clearly marked on equipment and floor layout drawings appended to each procedure. Any heavy-load-handling operation, prior to movement through an exclusion area, will be required by administrative control to undergo a plant engineering safety review and evaluation. Analyses have shown that the floor structure will withstand the impact of heavy load drops in safe load path areas where safe shutdown or decay heat removal equipment may lie below the floor structure. Based on the above, the applicant feels that marking the floors is unnecessary and impractical [5]. The applicant identified those heavy operations over or near irradiated fuel, reactor vessel, spent-fuel storage pool, or safe shutdown equipment and identifies those cases for which a load drop analysis will be performed [4].

B. EG&G Evaluation

The applicant provided detailed and well-illustrated drawings of the load paths for each overhead handling system and stated that the load paths were generally defined in accordance with the NUREG guidelines. LP&L stated that the load paths and exclusion areas will be defined and clearly marked on each load-handling procedure, and safety review is required for any deviations. EG&G concludes that adequate measures have been taken to ensure that load-handling operations remain within safe load paths.

C. EG&G Conclusions and Recommendations

Waterford Generating Station, Unit 3 is consistent with the criteria of NUREG-0612, Guideline 1, Safe Load Paths.

2.3.2 Load-Handling Procedures [Guideline 2, NUREG-0612, Article 5.1.1(2)]

"Procedures should be developed to cover load-handling operations for heavy loads that are or could be handled over or in proximity to irradiated fuel or safe shutdown equipment. At a minimum, procedures should cover handling of those loads listed in Table 3-1 of NUREG-0612. These procedures should include: identification of required equipment; inspections and acceptance criteria required before movement of load; the steps and proper sequence to be followed in handling the load; defining the safe path; and other special precautions."

A. Summary of Applicant's Statements

"Prior to Fuel Load, procedures will be developed and implemented to cover load-handling operations for heavy loads that are or could be handled over or in proximity to irradiated fuel or safe shutdown equipment. These procedures will include: identification of required equipment; inspections and acceptance criteria required before movement of load; the steps and proper sequence to be followed in handling the load; defining the safe path areas; and special precautions; if necessary [5]."

B. EG&G Evaluation

On the basis of the applicant's statement, EG&G feels that the criteria of NUREG-0612, Guideline 2 will be satisfied.

C. EG&G Conclusions and Recommendations

EG&G concludes that Waterford Generating Station, Unit 3 is consistent with the intent of criteria of NUREG-0612, Guideline 2, Load-Handling Procedures.

2.3.3 Crane Operator Training [Guideline 3, NUREG-0612, Article 5.1.1(3)]

"Crane operators should be trained, qualified, and conduct themselves in accordance with Chapter 2-3 of ANSI B30.2-1976, 'Overhead and Gantry Cranes' [6]."

A. Summary of Applicant's Statements

"LP&L has trained and qualified crane operators in accordance with Chapter 2-3 of ANSI B30.2-1976 [5]."

B. EG&G Evaluation

On the basis of the applicant's statement, EG&G concludes that the criteria of NUREG-0612 Guideline 3 has been satisfied. Training and qualification records must be made available for audit.

C. EG&G Conclusions and Recommendations

EG&G concludes that the Waterford Generating Station, Unit 3 is consistent with the intent of NUREG-0612, Guideline 3, Crane Operator Training.

2.3.4 Special Lifting Devices [Guideline 4, NUREG-0612, Article 5.1.1(4)]

"Special lifting devices should satisfy the guidelines of ANSI N14.6-1978, 'Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More for Nuclear Materials' [7]. This standard should apply to all special lifting devices which carry heavy loads in areas as defined above. For operating plants, certain inspections and load tests may be accepted in lieu of certain material requirements in the standard. In addition, the stress design factor stated in Section 3.2.1.1 of ANSI N14.6 should be based on the combined maximum static and dynamic loads that could be imparted on the handling device based on characteristics of the crane which will be used. This is in lieu of the guideline in Section 3.2.1.1 of ANSI N14.6 which bases the stress design factor on only the weight (static load) or the load and of the intervening components of the special handling device."

A. Summary of Applicant's Statements

The applicant identified six special lift devices that are to be used and discussed their evaluation as follows [5]:

1. Two of these devices (Disposable Cask Liner Lift Rig and Shipping Cask Lift Rig) were excluded from further consideration because they are designated for use on a monorail/hoist that has been excluded from further consideration because of physical separation by distance or limited load path. [5]

2. The Reactor Coolant Motor Lift Rig complies with the stress design factors addressed in ANSI N14.6-1978, Section 3.2.1.1, as supplemented by NUREG-0612, Section 5.1.1.(4). In addition, an analysis for a postulated drop of the RC pump motor to the operating floor elevation -11 ft was performed and its consequent effects were found acceptable. [5]

3. A heavy load drop analysis, prepared for the Core Support Barrel, indicates that the local and overall effects of the impact on the structure are acceptable. The analysis also determined that the travel path is not over any irradiated fuel and that the effects of a postulated drop of the Core Support Barrel are less critical than that of the upper guide structure. No postulated load drop was initiated for the upper guide structure since its effect is less critical than that of the vessel head. The CSB lift rig is part of the UGS lift rig and this device was evaluated with regard to the design and fabrication compliance with NUREG-0612 and ANSI N14.6-1978 criteria. Both lift rigs exceed NUREG-0612 stress allowances in a number of locations and do not fully meet all ANSI N14.6-1978 requirements. [5]

   Although the UGS and CSB lift rigs were not designed to ANSI N14.6, they were designed to approved standards and fabricated to stringent quality control and quality assurance procedures. The stress resulting from a design load of twice the operating load will not exceed the code allowable stress for the material of each load carr ing member. The code stress is the tensile or compressive stress allowed by the ASME Boiler and Pressure Vessel Code, Section III NB-3000 tables. This stress is always less than the minimum yield strength corresponding to a member's material specification [e.g. forged 304 SS has a code (allowable) stress about 1/3 less than minimum yield]. As discussed with the vendor engineers, this design criteria will result in a material yield to normal operating stress ratio (safety factor) of greater than 3 for most members of the subject lift rigs. For the remainder, the safety factor will be between 2.0 and 3.0 with most of these cases being closer to 3.0. It is LP &L Co.'s opinion that the margin to material yield and ultimate strength for the "low usage" CSB and UGS lifting rigs is comparable to the ANSI N14.6 margins for "high usage" lifting devices for nuclear material shipping containers. However, since verbatim compliance is not achievable in this case, appropriate non-destructive examination will be incorporated into our inservice inspection program. The UGS and CSB lift rigs have been subjected to a manufacturer recommended overload test of 125% as proof of workmanship prior to shipment to the site. The use of the detailed vendor provided Reactor Internals Lift Rig Manual assures that error is highly unlikely in any required assembly or disassembly. The lift devices are relatively uncomplicated and the number of weld joints have been minimized by the use of pin and bolt connectors to couple the tie rod, spreader and lift column assemblies. In addition to these design considerations, maintenance, repair and testing procedures will also insure a continued level of substantial safety margin throughout the useful lifetime of the rigs. Control identification and work authorization procedures establish the control condition which assures that repair work or replacement part orders will meet or exceed the original design criteria. LP&L Co. will examine all load bearing welds over a normal inservice inspection interval in a manner similar to that specified for ASME B&PV Code for Class 2 component supports. [10]

4. LP&L Co. prepared an item by item comparison of the RV head lift Rig to the ANSI N14.6 standard. The results of the comparison study are similar to those of the UGS and CSB lift rigs. Likewise the conclusions drawn for the UGS and CSB fixtures apply equally to the RV head Lift Rig. The stress safety factors meet or exceed ANSI N14.6 requirements for material yield and ultimate strength for all load carrying members of the lead lift rig with few exceptions. In these few cases, the safety margins for material yield are very close to ANSI N14.6. LP&L Co. will perform NDE required for Class 2 component supports for all load bearing welds over a 10 year ISI interval to insure these stress margins are maintained throughout the useful life of the RV Head Lift Rig. [10]

   The ISI interval testing of the UGS, CSB, and RV Head Lift Rigs will consist of visual inspection for weld free components and surface examination for integrally welded support members. [10]

B. <u>EG&G Evaluation</u>

On the basis of the information submitted,

1. EG&G agrees that the Disposable Cask Liner Lift Rig and the Shipping Cask Lift Rig may be excluded from further consideration.

2. EG&G concludes that the Reactor Coolant Motor Lift Rig is in compliance.

3. The applicant provided in Reference [5], an item by item comparison of the UGS and CSB Lift Rigs to the requirements of ANSI N14.6 and also provided sketches of these fixtures showing areas where stress levels exceed the NUREG-0612 allowables. The comparison did not demonstrate complete compliance with ANSI N14.6 requirements, however, additional information (Reference 10) provided by the applicant indicates reasonable assurance that sound engineering practices and Quality Assurance measures were employed in the design, fabrication and examination of the lift rigs. In service inspection also appears to be consistent with the intent of NUREG-0612. Even though stress levels do exceed NUREG-0612 allowables in some cases, EG&G Idaho does agree that with a disciplined controlled maintenance, repair, and testing program, margins of safety provided are adequate for the intended use.

4. LP&L Co. prepared an item by item comparison of the RV Head Lift Rig to requirements of ANSI N14.6. Although the comparison did not demonstrate complete compliance, the results do provide reasonable assurance that sound Engineering Practices and Quality Assurance measures were employed in the design, fabrication, and examination of the lift Rig. LP&L Co. also provided a stress summary that identifies the stresses for critical elements of the RV Head Lift fixture for an operating load that includes both static and dynamic loads. Although stress levels do not meet NUREG-0612 allowables for all elements of the fixtures, the factors of safety exceed 2.6 based on yield strength and 5.5 based on ultimate strength in all cases.

On the basis of this information provided, EG&G concludes that the UGS. CSB, an RV Head Lift Fixtures were provided by a Reactor Vendor in accordance with Quality Assurance and Quality Control procedures that appear to be appropriate for the specific application associated with handling of the components provided by the vendor. Although a 150% overload test has not been performed, the lifting devices have been subjected to the manufacturers recommended load test of 125% to demonstrate proof of workmanship.

C. EG&G Conclusions and Recommendations

EG&G concludes that Waterford Generating Station, Unit 3 is consistent with the intent of NUREG-0612, Guideline 4, Special Lift Devices.

2.3.5 Lifting Devices (Not Specially Designed) [Guideline 5, NUREG-0612, Article 5.1.1(5)]

"Lifting devices that are not specially designed should be installed and used in accordance with the guidelines of ANSI B30.9-1971, 'Slings' [8]. However, in selecting the proper sling, the load used should be the sum of the static and maximum dynamic load. The rating identified on the sling should be in terms of the 'static load' which produces the maximum static and dynamic load. Where this restricts slings to use on only certain cranes, the slings should be clearly marked as to the cranes with which they may be used."

A. Summary of Applicant's Statements

"A review of other lifting devices used in Waterford 3 including ropes, slings, and cables, will be done to determine the extent that the design, fabrication, and proof-testing methods used comply with the guidelines of ANSI B30.9-1971, as supplemented by NUREG-0612, Section 5.1.1.(5).

"In selecting the proper sling where the load is based on a combination of static and dynamic loads, the dynamic contribution of the rated load is taken as 1/2% (sic) of hoisting speed in feet per minute (fpm), but not less than 15%, nor more than 50% of the rated load. The hoisting speeds at Waterford 3 do not exceed 30 fpm. Hence, the dynamic contribution is 15%. While LP&L does not agree that dynamic loads must be addressed, the safety factor of 5 required by ANSI B30.9 is considered adequate to account for any required dynamic effect. This is ... strains (i.e., ... blocks). Additionally, if compliance with the above cannot be verified for a particular sling, then the sling will be load-tested to demonstrate its equivalency in terms of load handling reliability, or the sling will be replaced with one which meets the guidelines [5]."

B. EG&G Evaluation

On the basis of the applicants statement, EG&G concludes that LP&L Co. has evaluated the routine potential dynamic loading and determined it to be a relatively small fraction (15%) of static load because of the relatively slow hoisting speed (less than 30 fpm). EG&G also concludes that LP&L Co. intends to utilize a dynamic factor of 15% of the operating load in their equipment selection, however, where compliance to the NUREG-0612 criteria cannot be verified, the sling will be load tested to demonstrate rel[illegible]y

C. EG&G Conclusions and Recommendations

EG&G concludes that Waterford Generating Station, Unit 3 is consistent with the intent of NUREG-0612, Guideline 5 Lifting Devices (not specially designed).

2.3.6 Cranes (Inspection, Testing, and Maintenance) [Guideline 6, NUREG-0612, Article 5.1.1(6)]

"The crane should be inspected, tested, and maintained in accordance with Chapter 2-2 of ANSI B30.2-1976, 'Overhead and Gantry Cranes,' with the exception that tests and inspections should be performed prior to use where it is not practical to meet the frequencies of ANSI B30.2 for periodic inspection and test, or where frequency of crane use is less than the specified inspection and test frequency (e.g., the polar crane inside a PWR containment may only be used every 12 to 18 months during refueling operations, and is generally not accessible during power operation. ANSI B30.2, however, calls for certain inspections to be performed daily or monthly. For such cranes having limited usage, the inspections, test, and maintenance should be performed prior to their use)."

A. Summary of Applicant's Statements

"All cranes concerned will be inspected, tested, and maintained in accordance with the guidelines of Chapter 2-2 of ANSI B30.2-1976, Overhead and Gantry Cranes, with the exception that tests and inspections will only be performed prior to their use when it is not practical to meet the frequencies of ANSI B30.2 for periodic inspection and test, or where the frequency of crane use is less than the specified inspection and test frequency, and where the requirements of the rated load tests do not conflict with safe handling practices [4]."

B. EG&G Evaluation

The applicant noted the possible conflict between the Rated Load Test (ANSI B30.2-1976, Section 2-2.2.2) and Industry safe handling practice. However, the applicant stated that they do not anticipate any such situation to exist. It should be further noted that the Rated Load Test should be conducted prior to initial use. EG&G is in agreement with LP&L's proposed program.

C. EG&G Conclusions and Recommendations

EG&G concludes that Waterford Generating Station, Unit 3 is consistent with the intent of the criteria of NUREG-0612, Guideline 6 Cranes (Inspection, Testing, and Maintenance).

2.3.7 Crane Design [Guideline 7, NUREG-0612, Article 5.1.1(7)]

"The crane should be designed to meet the applicable criteria and guidelines of Chapter 2-1 of ANSI B30.2-1976, 'Overhead and Gantry Cranes,' and of CMAA-70, 'Specifications for Electric Overhead Traveling Cranes' [9]. An alternative to a specification in ANSI B30.2 or CMAA-70 may be accepted in lieu of specific compliance if the intent of the specification is satisfied."

A. Summary of Applicant's Statements

All cranes were designed, fabricated, installed, and tested in accordance with Ebasco specification which generally complies with the guidelines of CMAA-70, "Specification for Electric Overhead Traveling Cranes" and Chapter 2-1 of ANSI B30.2-1976, "Overhead and Gantry Cranes" or better [4]. A comparison was made for selected pertinent items between the Ebasco specification and CMAA-70. The applicant concluded that cranes furnished through the Ebasco specification in conjunction with CMAA-70 definitely satisfy the intent of either CMAA-70 and/or Chapter 2-1 of ANSI B30.2-1976 or better [5].

B. EG&G Evaluation

On the basis of the applicant's submittal, EG&G concludes that the Ebasco specification is more stringent that CMAA-70 for the selected pertinent items. Since CMAA-70 (1.8.1) invokes the safety features of ANSI B30.2.0 safety code and the applicant stated that the cranes were designed fabricated, installed, and tested in accordance with CMAA-70 or Ebasco specifications, whichever is more stringent [5], it must be concluded that the cranes also meet the requirements of ANSI B30.2. Procurement documents and specifications should be made available for audit.

C. <u>EG&G Conclusions and Recommendations</u>

EG&G concludes that Waterford Generating Station, Unit 3 is consistent with the criteria of NUREG-0612, Guideline 7, Crane Design.

2.4 Interim Protection Measures

The NRC staff has established (NUREG-0612, Article 5.3) that six measures should be initiated to provide reasonable assurance that handling of heavy loads will be performed in a safe manner until final implementation of the general guidelines of NUREG-0612, Article 5.1, is complete. Four of these six interim measures consist of general Guideline 1, Safe Load Paths; Guideline 2, Load-Handling Procedures; Guideline 3, Crane Operator Training; and Guideline 6, Cranes (Inspection, Testing, and Maintenance). The two remaining interim measures cover the following criteria:

- o Heavy load technical specifications

- o Special review for heavy loads handled over the core.

However, because the WGS No. 3 plant is currently not an operating facility nor will it be operating in the near future, EG&G recommends that LP&L not spend time and effort addressing the interim protection phase of NUREG-0612, but instead devote its efforts towards the completion of operating procedures and qualifications.

3. CONCLUDING SUMMARY

3.1 Applicable Load-Handling Systems

The list of cranes and hoists supplied by the applicant as being subject to the provisions of NUREG-0612 appears to be complete (see Section 2.2.1). The applicant has fulfilled the requirements of NUREG-0612 concerning exclusion of various overhead handling systems.

3.2 Guideline Recommendations

Waterford Generating Station Unit 3 has adequately demonstrated consistency with the seven NRC guidelines for heavy load handling (Section 2.3). This conclusion is represented in tabular form as Table 3.1. No further specific recommendations to aid in compliance with the intent of these guidelines are provided.

TABLE 3.1. COMPLIANCE MATRIX WATERFORD GENERATING STATION UNIT 3

Equipment Designation	Heavy Loads	Capacity (Tons)	Guideline 1 Safe Load Paths	Guideline 2 Procedures	Guideline 3 Crane Operator Training	Guideline 4 Special Lift Devices	Guideline 5 Slings	Guideline 6 Crane-Test and Inspect	Guideline 7 Crane Design
Reactor Cont. Building Polar Crane	C	200/30	C	C	C	C	C	C	C
Fuel-Handling Building	C	125/15	C	C	C	C	C	C	C

C = pplicant action fully complies with NUREG-0612 Guideline, subject to review by NRC Staff.

NC = Applicant action doe not fully comply with NUREG-0612 Guideline, subject to review by NRC Staff.

4. REFERENCES

1. NUREG-0612, Control of Heavy Loads at Nuclear Power Plants, NRC.

2. V. Stello, Jr. (NRC), Letter to all applicants. Subject: Request for Additional Information on Control of Heavy Loads Near Spent Fuel, NRC, May 17, 1978.

3. USNRC, Letter to LP&L Co. Subject: NRC Request for Additional Information on Control of Heavy Loads Near Spent Fuel, NRC, December 22, 1980.

4. Letter to NRC; Subject: Waterford 3 SES Control of Heavy Loads, from L. V. Maurin, Assoc. Vice Pres. LP&L Co. to D. G. Eisenhut, Director Division of Licensing, USNRC, dated June 19, 1981.

5. Letter to NRC; Subject: Response to EG&G Draft Tech Evaluation Report, from L. V. Maurin, LP&L Co. dated January 27, 1983.

6. ANSI B30.2-1976, "Overhead and Gantry Cranes."

7. ANSI N14.6-1978, "Standard for Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or more for Nuclear Materials."

8. ANSI B30.9-1971, "Slings."

9. CMAA-70, "Specifications for Electric Overhead Traveling Cranes."

10. Letter to NRC; Subject: Waterford SES Unit 3 Control of Heavy Loads--Special Lifting Devices, from K. W. Cook, Nuclear Support and Licensing Mgr LP&L Co. to G. W. Knighton Division of Licensing USNRC, dated February 9, 1984.

11. Documentation of Telephone Communication, Kevin Curley LP&L Co. to J. P. Sekot, EG&G Idaho, September 22, 1983.

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Docket No. 50-382

13. ABSTRACT (200 words or less)

Supplement 6 to the Safety Evaluation Report for the application filed by Louisiana Power & Light Company for a license to operate the Waterford Steam Electric Station, Unit 3 (Docket No. 50-382), located in St. Charles Parish, Louisiana, has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The purpose of this supplement is to update the Safety Evaluation Report by providing the staff's evaluation of information submitted by the applicant since the Safety Evaluation Report and its five previous supplements were issued.

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