SEP 24 1984

Docket No.: STN 50-483

Mr. D. F. Schnell
Vice President - Nuclear
Union Electric Company
P. O. Box 148
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Dear Mr. Schnell:

Subject: NRC Staff Evaluations for the Callaway Plant, Unit 1

Enclosed are copies of the staff's evaluation for the Callaway Plant, Unit 1 which the staff proposes to incorporate into the next SER supplement. The enclosed draft reports provide staff review of those items that required resolution prior to exceeding 5% of rated power and address changes to the SER which resulted from the receipt of additional information. If there are any questions concerning the enclosed reports, please contact Mr. Joseph Holonich, the Callaway Project Manager.

Sincerely,

/s/

B. J. Youngblood, Chief
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Division of Licensing

Enclosure:
As stated

cc: See next page

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9/10/84	9/20/84	9/ /84

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2 SITE CHARACTERISTICS

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2.4 Hydrology

2.4.4 Ultimate Heat Sink

In the SER the staff stated that an independent analysis of the thermal and hydrologic performance of the essential service water system was not made because of the significant margin available in the volume of the ultimate heat sink (UHS) retention pond over the requirements for one unit. To retain this margin, the staff established the UHS minimum water depth requirement in the final draft ("Technical Specifications for Callaway Unit No. 1") at 16 ft above the pond bottom. The licensee, however, requested a lower minimum water depth in the pond so as to eliminate spillway discharges during tests and normal operation. Such spillway discharges would be in violation of the licensee's National Pollution Discharge Elimination System permit for one-unit operation.

As an alternative to the 16-ft minimum depth established by the staff, the licensee proposed a minimum water depth of 13.25 ft (May 23, 1984). This depth would contain the water required for 30 days of losses under severe meteorological conditions plus a margin of 50% over the calculated loses. The staff reviewed the licensee's proposed water depth and found it acceptable. The UHS Technical Specification has been changed to reflect the licensee's proposed water level.

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3 DESIGN CRITERIA FOR STRUCTURES, SYSTEMS, AND COMPONENTS

3.10 Seismic and Dynamic Qualification of Safety-Related Mechanical and Electrical Equipment

3.10.1 Seismic and Dynamic Qualification

As discussed in SSER 3, the staff's evaluation 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, standards followed, and the completeness of the program in general and (2) an audit of selected equipment items to develop the basis for staff judgment on the completeness and adequacy of the implementation of the entire seismic and dynamic qualification program. The Seismic Qualification Review Team (SQRT) consisting of staff engineers and engineers from the Idaho National Engineering Laboratory (INEL) reviewed the equipment dynamic qualification information in FSAR Sections 3.9.2 and 3.10 and visited the plant site on December 5 through December 7, 1984, to determine the extent to which the qualification of equipment as installed at SNUPPS plants meets the current licensing criteria described in Regulatory Guides 1.100 and 1.92, SRP Section 3.10, and Institute of Electrical and Electronics Engineers (IEEE) Std. 344-1975. Conformance with these criteria is required to satisfy the applicable portions of GDC 1, 2, 4, 14, and 30 (Appendix A to 10 CFR 50), Appendix B to 10 CFR 50, and Appendix A to 10 CFR 100.

Discussion of the initial results of the SQRT findings and review of information submitted by the licensee, including justification for interim operation up to 5% power operation, can be found in SSER 3. Since issuance of SSER 3, the staff has completed its review of additional information submitted by the licensee. On the basis of the audit and review of the licensee's submittals, it is the staff's opinion that the SNUPPS seismic and dynamic qualification of equipment program has been satisfactorily defined and implemented to the current staff criteria as stated above.

The staff's findings are summarized in Sections 3.10.1.1, 3.10.1.2, and 3.10.1.3 of this report, and a summary of the staff' evaluation of the applicant's program is provided in Section 3.10.1.4.

3.10.1.1 Generic Issues

As stated in SSER 3, all the generic issues were resolved.

3.10.1.2 Specific Issues

The status of equipment-specific issues remains the same as stated in SSER 3.

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3.10.1.3 Justification for Interim Operation

As discussed in SSER 3, eleven categories of equipment were not specifically included among the items reviewed by the SQRT and whose qualification was not expected to be fully completed before low-power operation. SNUPPS provided adequate justification for interim operation (JIO), which, in the opinion of the staff, was adequate for 5% power operation.

Subsequently, SNUPPS submitted additional JIO, in its letters of June 29 and July 16, 1984, to its request for full-power operation while the qualification program for some of these equipment items is in progress. As is indicated, the licensee has, in some cases, incorporated the JIO in the equipment qualification documentation. In all such cases, the licensee has stated that testing has been successfully completed according to the staff licensing criteria. When formal documentation is available, it will be substituted for the JIO in the documentation file to ensure uniformity of the file.

The staff reviewed the additional information as provided in the above SNUPPS letters and found that some of the previously unqualified equipment items have now been completely qualified for SNUPPS application, and that the associated JIO as discussed in SSER 3 should be terminated. The staff has also found that the additional justifications for interim operation, as presented for other equipment items, are acceptable to the staff for supporting full-power operation of SNUPPS plants. Discussion for each individual equipment item follows.

Crosby Position Indication Device (HE-7)

On the basis of previous tests, discussed in SSER 3, the failure mechanism of the position indication device (PID) had been concluded to be the moisture/chemical spray inwicking along the lead wires that damaged reed switches and degraded electrical performance of the switches. The licensee was committed to have the connection sealed with seismically and environmentally qualified Conax connectors. In addition, previous seismic testing has provided acceptable evidence that the PID is seismically qualified. On the basis of the above and the fact that the complete qualification test of the assembly of the individually qualified items is in process and will be completed by December 1984, the staff concludes that the SNUPPS JIO is acceptable for full-power operation. The licensee should provide a written confirmation that the qualification test, when completed, will meet the regulatory requirements.

7300 Process Protection System (ESE-13)

As discussed in SSER 3, the licensee was committed to complete the qualification program for this item before exceeding 5% power operation. Review of the JIO for this system has led to a conclusion that the equipment in the Callaway Plant is seismically qualified. The JIO will be used as supporting documentation for seismic qualification until the final documentation is finished. Final documentation is required to ensure uniformity of data in the equipment qualification document files and will be completed by March 1985. The licensee should provide a written confirmation that the qualification program, when completed, will meet the regulatory requirements.

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Boron Dilution Protection System (ESE-47)

As stated in SSER 3, the operational concern on source-range preamplifier, a part of the system, leads to a new preamplifier (model MK II) to replace the old one (model MK I). Functionally, testing has proven this to be a superior design. However, the new redesigned triaxial connector, which was in the field, failed during the seismic test. The old style connector was then installed and subsequent seismic test results on the preamplifier were satisfactory. Furthermore, the licensee has already placed these old connectors in the field.

On the basis of the above, the seismic qualification of the ESE-47 equipment has been demonstrated for SNUPPS application, and the staff is in agreement that the JIO of this equipment should be terminated.

Thermocouple/Core Cooling Monitor System (ESE-56A)

The failure of the plasma display during safe shutdown earthquake (SSE) testing in positions 3 and 4 is attributed to fretting of the edge connector contacts and board edge fingers which produces microscopic particles of oxidized material that act as an insulator causing intermittent open circuits. On the basis of the symmetry of construction, the direction of excitation is determined to be an insignificant factor in fretting. It is, therefore, concluded that the unit is adequate for one SSE. To provide additional margin, however, the manufacturer was developing a lubricant/oxidation inhibitor which would be applied at SNUPPS. As a result of a later decision by the licensee, the inhibitor will not be applied.

The problem of intermittent output from the PS-2 power supply during seismic testing was attributed to temperatures greater than or equal to 138°F. This was confirmed when performance resumed after the temperature was reduced. For SNUPPS, this system is located in the control room which has Class 1E heating, ventilation, and air conditioning (HVAC) and will not likely experience abnormal temperature. However, further testing is scheduled to qualify the PS-2 power supply for different, harsh environment applications. The TC/CCM system has been successfully seismically tested after certain hardware modifications. The seismic qualification of this system will be considered demonstrated when Westinghouse Field Change Notice (FCN) SCPM-10622 has been completed for Callaway. The licensee has committed to complete the FCNs before exceeding 5% power. The staff finds the JIO acceptable and it will serve as documentation of qualification for the system until formal documentation, scheduled for completion in November 1984, is available. The staff will ensure that the field modifications are completed. In the meantime, the licensee should provide a written confirmation that the qualification program, when completed, will meet the regulatory requirements.

International Instruments Model 1151 Indicators (J-110)

Adequate seismic testing has been performed for SNUPPS by American Environments and witnessed by Bechtel Power Corporation. Minor anomalies which occurred were judged to be insignificant. The qualification program, including full documentation, has been completed. The staff agrees that the JIO of this equipment should be terminated.

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AWV Model 7401 Dampers (M-627A)

A seismic test to verify the acceptability of the modified dampers for SNUPPS was completed in February 1984. The results were determined to be satisfactory. Test reports have been reviewed and approved. The dampers, therefore, have been fully qualified for SNUPPS applications. The staff agrees that the JIO of this equipment should be terminated.

Operator Interface Module (ESE-12A)

The meters were required to demonstrate a combined worst-case accuracy of ±5.5% of calibrated span during the seismic and abnormal environment testing. The switches must demonstrate absence of contact bounce during seismic testing. Potentiometers and switches must function before and after each event, but not during.

During seismic testing, all four current meters were well within accuracy requirements, and the meters are qualified with no anomalies of the associated switches observed. One of the three brush recordings for the 500-ohm potentiometer, however, indicated momentary interruptions of the signal. Such anomaly is not significant since, as stated above, the potentiometer is not required to function during the event.

The anomaly of the current meters, which was observed during abnormal environment testing at high temperatures, is not applicable to the SNUPPS plants because of the SNUPPS Class 1E control room HVAC systems.

On the basis of the above discussion, the staff agrees that seismic qualification of the ESE-12A equipment for SNUPPS applications has been demonstrated, and the JIO should be terminated.

Cutler Hammer Series E-30 Pushbutton Assemblies (E-028, J-200)

The seismic testing of E-30 pushbutton assemblies has been completed at Wyle Laboratories. Testing was performed to the requirements of IEEE Std. 323-1974 and IEEE Std. 344-1975. The qualification program, therefore, has been completed. The staff agrees that the JIO of this equipment should be terminated.

Head Vent System Control Module (HE-10B)

A seismic test of this module, utilizing multiaxis, multifrequency input, has been performed which met or exceeded the prescribed requirements; no failures were detected. The qualification program, including full documentation, has been completed. The staff agrees that the corresponding JIO should be terminated.

Incore Thermocouples, Connectors, Adapters, and Junction Box--Core Cooling Monitor System (ESE-43 and ESE-44)

The JIO was based on a nearly completed qualification test series with evidence that the series could be successfully completed. The testing of the junction box for postaccident radiation exposure needed repeating because of a loss of seal on the original loss-of-coolant accident (LOCA) testing. This did not affect the seismic qualification of the box because the occurrence of a seismic

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event following a design-basis accident has not been defined as a credible event. The JIO which describes seismic testing is considered acceptable for the SNUPPS equipment. Formal documentation is scheduled to replace the JIO in the documentation file in December 1984. At that time, the licensee should provide a written confirmation that the qualification program, when completed, will meet the regulatory requirements.

Barton Differential Pressure Indicating Switches (ESE-40) Model Nos. 288A and 681A

The JIO was based on previously completed testing and an analysis indicating that seismically induced chatter, shown to be possible by the testing, will not degrade the performance of the systems in which the switches are installed to unacceptable levels. A change in switch setpoint in the field is required to ensure this. The licensee has proposed to make the setpoint adjustment before exceeding 5% power and to ensure that such adjustment will not invalidate conformance of the previous test results to IEEE Std. 344-1975. This JIO is acceptable to the staff and will serve as an interim documentation. Full documentation will be completed in December 1984. At that time, the licensee should provide a written confirmation that the qualification program, when completed, will meet the regulatory requirements.

3.10.1.4 Summary

On the basis of SQRT audit findings as well as on the review of subsequent submittals, including the justification for interim operation, the staff concludes that an appropriate seismic and dynamic qualification program has been defined and implemented which provides adequate assurance that such equipment should function properly during and after the excitation from vibratory forces imposed by the SSE. The staff finds that the SNUPPS seismic and dynamic qualification program is acceptable.

On the basis of the staff review and acceptance of the justification for interim operation and the staff requirement that the licensee provide written confirmation of the completion of all items of the seismic and dynamic qualification program in accordance with approved standards, the staff recommends full-power operation for Callaway, Unit 1.

3.10.2 Operability Qualification of Pumps and Valves

As discussed in SSER 3, the staff performed a two-step review to ensure that the licensee has provided an adequate program for qualifying safety-related pumps and valves to operate under normal and accident conditions. The first step was a review of FSAR Section 3.9.3.2 for the description of the licensee's pump and valve operability assurance program. The second step involved an onsite audit of a small representative sample of safety-related pumps and valves and supporting documentation by the Pump and Valve Operability Review Team (PVORT).

The two-step review was performed to determine the extent to which the qualification of equipment, as installed, meets the current licensing criteria in SRP Section 3.10. Conformance with these criteria provides an acceptable way of meeting the applicable portions of GDC 1, 2, 4, 14, and 30 as well as Appendix B to 10 CFR 50.

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During the PVORT review, some concerns were raised. The licensee resolved all of the major specific concerns during the audit, either by supplying additional information or demonstrating that the appropriate commitments are already addressed by administrative controls. However, the staff requested confirmation of a few items to resolve staff concerns as discussed in SSER 3. The following is a discussion of the resolution of those items.

3.10.2.1 Generic Findings

In SSER 3, the staff required that the SNUPPS FSAR Section 3.9.3.2 be amended to provide a more current and detailed description of the pump and valve operability program, including a description of the criteria for determining which balance-of-plant (BOP) and nuclear-steam-supply-system (NSSS) pump and valve accessories are incorporated into the FSAR lists of active safety-related equipment. By Letter SLNRC84-0045 and 84-0086, dated March 16 and March 24, 1984, respectively, the licensee committed to comply with the staff request in a future revision of the SNUPPS FSAR. This response is acceptable to the staff.

The staff also required the licensee to verify that all safety-related equipment is fully qualified, and the licensee addressed this in Letters 84-0045 and 84-00101 dated March 16 and June 29, 1984. The recipient's and subject's staff reviewed these responses and concluded that except for equipment-specific issues which are discussed below in Section 3.10.2.2, all generic concerns are resolved.

3.10.2.2 Equipment-Specific Issues

There are qualification programs for equipment affecting pumps and valves, which are not expected to be completed before 5% power is exceeded at Callaway Unit 1. However, the applicant provided justification for interim operation (JIO) in April 1984, which the staff reviewed and considered acceptable to operate Callaway Unit 1 at a 5% power level. The staff reviewed the subject justification and found it satisfactory, because the licensee had (1) presented a rigorous test program based on methodologies in conformance with IEEE Stds. 323-1974 and 344-1975 and Regulatory Guides (RGs) 1.89, 1.100, and 1.73; (2) established maintenance programs in conformance with RG 1.33 to ensure that the equipment is maintained in a qualified status throughout the plant life; and (3) committed to complete qualification no later than March 1985.

On June 29, 1984, the licensee provided additional information regarding the justification of interim operation in order to justify plant operation above the 5%-power level. The staff reviewed the latest submittal. The staff's review and acceptance are based on the following reasons.

(1) JIOs HE-1, HE-9, HE-102, and HE-106 were issued because the documentation has not been reviewed in accordance with the SNUPPS procedures described in the SNUPPS's submittal for NUREG-0588. The JIOs, which are standard Westinghouse Equipment Qualification Data Packages (EQDPs), document the successful completion of rigorous testing programs to the requirements of IEEE Stds. 323-1974 and 344-1975.

(2) The licensee has stated that the above equipment does comply with the operability qualification provisions of the SNUPPS FSAR, although the SNUPPS review of the documentation is not complete. The licensee has committed to complete its review by March 1985.

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(3) The equipment accessories, whose qualification is incomplete, impact the safety function of the system minimally. JIO HE-7 addresses the qualification of a position-indication device, the design of which is such that it does not cause malfunction of the pressurizer safety valve. JIO J-601A addresses the qualification of NAMCO limit switch for a design-basis accident (DBA) radiation. The associated containment isolation valve will perform its safety function within minutes of the beginning of the DBA. Any subsequent failure of the limit switch will not cause the valve to change position. The licensee has committed to close out JIO HE-7 and J-601A by December 1984 and March 1985, respectively.

The staff, however, requires that the licensee, upon completion of the qualification program based on methodology accepted by the staff, confirms in writing that the program, including upgrading of equipment qualification files, is complete and that the governing qualification standards are met.

On the basis of the results of the site review performed for Callaway Unit 1 between December 5 and 7, 1983, and the subsequent submittals by the licensee to resolve issues identified from the site review, the staff has concluded that an appropriate pump and valve operability qualification program has been defined and implemented. The staff finds that the SNUPPS pump and valve operability assurance program is acceptable.

On the basis of the staff review and acceptance of the justification of interim operation and the requirement of written confirmation by the licensee of the completion of all items of the pump and valve operability qualification program in accordance with approved standards, the staff recommends full-power operation for Callaway Unit 1.

3.11 Environmental Qualification of Safety-Related Electrical Equipment

Section 3.11 of Callaway SSER 3 listed three license conditions which were made part of the Callaway operating license. License Condition 2.C.(3)(a) required that specification M723, "Seal Water Injection Filter," must be qualified before 5% of rated power is exceeded.

By letter dated June 29, 1984, the licensee stated that further analysis has been performed and the seal water injection filter is now considered qualified for its intended function. On the basis of information provided in the above letter, the staff finds that License Condition 2.C.(3)(a) has been satisfied and may be removed from the Callaway operating license.

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5 REACTOR COOLANT SYSTEM

5.2 Integrity of Reactor Coolant Pressure Boundary

5.2.4 Preservice and Inservice Inspection and Testing of the Reactor Coolant Pressure Boundary

This section was prepared with the technical assistance of U.S. Department of Energy (DOE) contractors from the Idaho National Engineering Laboratory.

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

This evaluation supplements conclusions in this section of the SER (NUREG-0830), which addressed the definition of examination requirements and the evaluation of compliance with 10 CFR 50.55a(g). The staff reviewed the selection of primary boundary welds subject to examination, as defined in the Callaway Preservice Inspection (PSI) Program, and found the sample selected for examination acceptable as reported in Supplement 3 (SSER 3)

In letters dated January 18, February 7, February 13, February 24, March 26, April 9, and June 13, 1984, the licensee requested relief from the ASME Code, Section XI requirements that had been determined to be impractical. These relief requests address the required volumetric examination of small-bore piping in the reactor coolant system, reactor pressure vessel examination, pressurizer examination, and random component and piping welds. The licensee provided supporting information pursuant to 10 CFR 50.55a(a)(2)(i). The staff evaluated the examinations required by the ASME Code that the applicant determined to be impractical and, pursuant to 10 CFR 50.55a(a)(2), has allowed relief from the impractical requirements, which, 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 specific preservice examination requirements, the staff concludes that the Callaway PSI Program meets the requirements of Section XI of the ASME Code, 1977 Edition, including Addenda through Summer 1978, and, therefore, is in compliance with 10 CFR 50.55a(g)(3). The detailed evaluation supporting this conclusion is provided in Appendix I to this report.

The initial inservice inspection program has not been submitted. This program will be evaluated after the applicable ASME Code edition and addenda can be determined on the basis of 10 CFR 50.55a(b), but before the first refueling outage when inservice inspection commences.

5.3 Reactor Vessel

5.3.1 Reactor Vessel Materials and Compliance With Appendices G and H, 10 CFR Part 50

In its SER, the staff indicated that an exemption to the uppershelf Charpy V-notch (CVN) impact energy requirements of Appendix G, 10 CFR 50, was necessary.

However, on July 26, 1984, Appendix G, 10 CFR 50, was revised. The revision permitted licensees to use materials that do not meet the upper-shelf requirements of Appendix G, 10 CFR 50, provided it is demonstrated in a manner approved by the Director, Office of Nuclear Reactor Regulation, that lower values provide margins of safety against fracture equivalent to those required by Appendix G of the ASME Code.

For the Callaway Plant, the staff evaluated the low upper-shelf material using the method of predicting radiation damage, which is documented in Regulatory Guide 1.99, Revision 1, "Effect of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." As a result of this evaluation, the staff has concluded that the material's CVN upper-shelf impact energy would remain above the safety margins required by Appendix G, 10 CFR 50, for more than 32 effective full-power years, which is the design life of the reactor vessel. As a result of the change to the regulation and previous approval of the Regulatory Guide by the NRC, an exemption is no longer required and additional approval is not necessary.

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6 ENGINEERED SAFETY FEATURES

6.2 Containment Systems

6.2.6 Containment Leakage Testing

Containment Air Lock Surveillance

By letter dated June 25, 1984, the licensee requested an exemption from the requirement of Paragraph III.D.2(b)(ii) of Appendix J to 10 CFR 50, which states: "Air locks open during period when containment integrity is not required at the end of such periods at not less than P_a."

The above Appendix J requirement would require a full-pressure air lock test after each and every shutdown regardless of the purpose of the shutdown. In lieu of this requirement, the licensee proposes to perform a full-pressure air lock test only when maintenance is performed on the air lock that could affect the sealing capability of the air lock. This proposed change requires an exemption from the requirements of Appendix J to 10 CFR 50. The staff's evaluation of this exemption request follows.

Whenever the plant is in cold shutdown (mode 5) or refueling (mode 6), containment integrity is not required. However, if an air lock is opened during modes 5 and 6, Paragraph III.D.2(b)(ii) of Appendix J requires that an overall air lock leakage test at not less than P_a be conducted before plant heatup and startup (i.e., entering mode 4). The existing air lock doors are so designed that a full-pressure (i.e., P_a (48.0 psig) test of an entire air lock can only be performed after strong backs (structural bracing) have been installed on the inner door. Strong backs are needed because the pressure exerted on the inner door during the test is in a direction opposite to that of the accident pressure direction. Installing strong backs, performing the test, and removing strong backs, require at least 6 hours per air lock, during which access through the air lock is prohibited.

If the periodic 6-month test in accordance with Paragraph III.D.2(b)(i) of Appendix J and the test required by Paragraph III.D.2(b)(iii) of Appendix J are current, no maintenance has been performed on the air lock, and the air lock is properly sealed, there should be no reason to expect the air lock to leak excessively just because it has been opened in mode 5 or mode 6.

Accordingly, the staff concludes that the applicant's proposed approach of substituting the seal leakage test of Paragraph III.D.2(b)(iii) for the full-pressure test of Paragraph III.D.2(b)(ii) of Appendix J is acceptable for Callaway Plant, Unit 1.

6.6 Inservice Inspection of Class 2 and 3 Components

This section was prepared with the technical assistance of DOE contractors from the Idaho National Engineering Laboratory.

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6.6.1 Evaluation of Compliance for Callaway Unit No. 1 With 10 CFR 50.55a(g)

This evaluation supplements the conclusions in this section of the SER, which addressed the definition of examination requirements and the evaluation of compliance with 10 CFR 50.55a(g). The staff reviewed the selection of the pressure boundary welds subject to examination, as defined in the Callaway PSI Program, and found the sample selected for examination acceptable, as reported in SSER 3.

In Letter ULNRC-839 dated June 3, 1984, the licensee identified a number of longitudinal seam pipe welds requiring preservice examination. These welds were not included in the PSI Program and a comprehensive evaluation recently indicated the need for extending the preservice examination effort using surface and/or volumetric methods to 80 additional longitudinal seam pipe welds located in the following systems:

(1) System EJ, residual heat removal--63 longitudinal seam welds

(2) System EM, high pressure coolant injection--16 longitudinal seam welds

(3) System EP, accumulator safety injection--1 longitudinal seam welds

The required field examinations have been satisfactorily completed and no problems were identified.

The staff has reviewed the licensee's letter dated June 3, 1984, describing additional longitudinal seam pipe welds requiring preservice examinations. An objective of preservice and inservice inspections is to systematically verify the as-built configuration in the region of the components required to be examined. This process was accomplished at the Callaway Plant where the examination personnel identified discrepancies in drawings that were reported to the licensee who took corrective action to determine the scope of the program and to perform all required examinations. In a letter dated June 13, 1984, the licensees revised the Callaway PSI Program to incorporate the pipe spools containing the additional longitudinal seam welds. The preservice examinations performed on the additional 80 welds include 59 welds which received surface examinations and 21 welds which received both volumetric and surface examination. The extent of the examination for all of the longitudinal seam welds was a region 2.5 times the pipe wall thickness measured from the intersecting circumferential weld.

On the basis of the review of the above information, the staff concludes that the licensee has identified all longitudinal seam pipe welds required to be examined and completed the preservice examinations on the basis of the requirements of the applicable editions of Section XI of the ASME Code.

In letters dated January 18, February 7, February 13, February 24, March 26, April 9, June 13, and June 27, 1984, the licensee requested relief from the ASME Code, Section XI requirements that had been determined to be not practical. These relief requests address the required volumetric examination of random component and piping welds and the visual examination of ASME Code, Class 3 supports. The licensee provided supporting information pursuant to 10 CFR 50.55a(a)(2)(i). The staff evaluated the examinations required by the ASME Code that the licensee determined to be impractical and, pursuant to 10 CFR 50.55a(a)(2), has allowed relief from the impractical requirements which if implemented, would result in

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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 specific preservice examination requirements, the staff concludes that the Callaway PSI Program meets the requirements of Section XI of the ASME Code, 1977 Edition including addenda through Summer 1978, and, therefore, is in compliance with 10 CFR 50.55a(g)(3). The detailed evaluation supporting this conclusion is provided in Appendix I to this report.

The initial inservice inspection program has not been submitted. This program will be evaluated after the applicable ASME Code edition and addenda can be determined on the basis of 10 CFR 50.55a(b), but before the first refueling outage when inservice inspection commences.

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7 INSTRUMENTATION AND CONTROLS

7.2 Reactor Trip System

7.2.2 Resolution of Issues

7.2.2.8 Environmental Errors for Reactor Trip Setpoints

By letter dated May 16, 1984, the staff requested that the licensee provide information before operation above 5% power or justify the omission of environmental errors for setpoint calculations related to the diverse trip functions or to incorporate appropriate environmental errors.

SNUPPS stated in a letter dated June 26, 1984, that for each event that could result in adverse environmental conditions, there is at least one actuation function available as a backup that is not located in the vicinity of the accident. Thus, it is not necessary to include environmental errors for setpoint calculations associated with such backup trips. The licensee did note, however, that if a trip function is diverse for one event but primary for another, the setpoint for both cases is based on the primary actuation function. Further, if a trip function is used in the safety analysis as a primary trip for an event, the actuation setpoint is based on the requirements of that event (i.e., if that event includes adverse environmental conditions in the vicinity of the sensor/transmitter, an environmental allowance is included). Also, the licensee reiterated that no credit is taken for the functioning of the diverse trip functions in the plant's FSAR accident analyses.

On the basis of the above discussion, the staff concludes that the licensee has provided sufficient information to justify the omission of environmental errors for setpoint calculations associated with the diverse trip functions. Thus, the staff concludes, with reasonable assurance, that the facility can be operated without undue risk to the health and safety of the public. This issue is considered resolved.

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9 AUXILIARY SYSTEMS

9.5 Other Auxiliary Systems

9.5.1 Fire Protection

9.5.1.5 Alternate Shutdown

In Section 9.5.1.5 of Supplemental Safety Evaluation Report (SSER) 3 the staff concluded that the alternative shutdown capability for the control room at the Callaway plant met the requirement of Branch Technical Position CMEB 9.5-1. This conclusion was based on staff review of (1) the final safety analysis report (FSAR) for standarized nuclear unit power plant systems (SNUPPS) and (2) the control room fire hazard analysis dated November 15, 1982, as well as the staff's understanding that all systems necessary to achieve and maintain hot shutdown could be isolated (which the staff assumed included operability) from the control room following fire damage to any circuits in the control room by placing the isolation switches (outside the control room) to the isolated position.

A recent inspection at Wolf Creek nuclear power plant revealed that in order to isolate some systems necessary for hot shutdown (other than those on alternate shutdown panel B) from control room fire damage and to maintain operability without replacing fuses, isolation must take place before fire damage occurs. Because Callaway and Wolf Creek are duplicate plants, this concern is also directly applicable to Callaway. Although the present isolation switches at SNUPPS plants do isolate the required equipment or components from the control room, it may be necessary to replace fuses as a result of control room fire damage, in order to place the equipment/component in the desired mode of operation or position. The alternate shutdown procedures used at Callaway are based on the assumption that the transfer switches will be placed in the isolated position before fire damage occurs in the control room that could result in fuse failure in the control power circuit. For such a case the isolation switches would isolate the desired component/equipment from the control room and operability would not be affected, since the fuses would now be isolated from the control room circuitry. At this point any further fire damage (hot short, open, or short to ground) would not affect the component(s) in question.

However, staff conclusions reached in SER Supplement 3 were based on the understanding that it would not be necessary to replace fuses after the transfer switches were placed in the isolated position, regardless of the time frame assumed for fire damage to the control room circuits. Following the inspection, the staff recognized that the present SNUPPS design in combination with the alternate shutdown procedures did not meet staff requirements for alternative shutdown capability in the event of a control room fire.

As a result of meetings with the SNUPPS utilities on August 10, 14, 15, and 22, 1984, the staff determined that new procedures could take care of many of the concerns identified by the inspection, since breakers or valves could still be operated locally. In other cases it was determined that the replacement of fuses was acceptable, since the components in question did not have an immediate effect on hot shutdown and ample time was available to replace fuses. However, there were four instances in which the licensee identified isolation switches that required modifications and five instances in which new isolation switches would have to be added. The new and modified isolation switches will have redundant fuses so that when placed in the isolation position new fuses would be switched into circuitry and the equipment would be isolated and immediately available.

By submittal dated August 23, 1984, the licensee provided a detailed outline of new alternate shutdown procedures and identified where the new and modified switches were required. The proposed new procedures consist of five phases, A through F, which will be performed by four operators. The new procedures assume that the control room is evacuated when the fire starts and operations outside the control room systematically bring all hot shutdown systems on the line and compensate for or prevent spurious operations that could affect achieving or maintaining hot shutdown.

Before the operator leaves the control room, he trips the reactor and closes the main steam isolation valves (MSIVs), if the fire permits him to do so. During phase A, which is completed within 5 minutes of evacuation, one operator establishes control at the alternate shutdown panel (ASP) using motor-driven pump B (after the diesel is running) and the atmospheric dump valves for steam generators B and D. The ASP operator also isolates the normal letdown path via an isolation switch on the ASP and closes the atmospheric dump valves for steam generators A and C. Meanwhile other operators simulate a loss of offsite power (if not lost), strip the loads from the 4160-B bus which is isolated from the effects of a control room fire, and start the diesel generator and essential service water (ESW) flow to the diesel generator. Also during phase A an operator trips the reactor coolant pumps if they are running, and isolates the power-operated relief valves (PORVs) via a knife switch. To ensure that spurious operation of atmospheric dump valves for steam generators A and C does not affect hot shutdown, an operator (during phase D) manually closes an isolation valve for each dump valve. New isolation switches will be added, to ensure that ESW valves HV-26 and HV-38 are properly positioned. HV-26 isolates the ESW system from the service water system and HV-38 is the ESW return to the ultimate heat sink (UHS). Until these switches are installed, an operator will trip the valve breakers (motor-operated valves) and will manually operate the valves if they need to be repositioned. Phase A will be completed within 5 minutes and at its completion (1) hot shutdown is being maintained at the ASP, (2) diesel generator B is running with cooling water being supplied by ESW train B, (3) the reactor coolant pumps (RCPs) are secured to protect the seals, and (4) some of the primary and secondary systems have been isolated (letdown, PORVs, and atmospheric dump valves). Although the turbine-driven AFW pump is isolated, it will not be used until an operator has assured that a suction flow path is available in phase D.

During phase B, which is completed within 10 minutes after the control room has been evacuated, operators maintain control at the alternate shutdown panel, verify turbine trip, initiate room cooling for the ESW pump room and the diesel generator room, and start the air conditioning systems for the control building and auxiliary building to ensure that vital electrical areas will be cooled. Also during phase B, the isolation valves between the refueling water storage tank (RWST) and the residual heat removal (RHR) pump suctions are closed to

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preclude the RWST from inadvertently draining to the containment recirculation sump. New/modified isolation switches will be provided for the ESW and diesel generator inlet dampers and supply fans to ensure timely initiation of room cooling for these areas. In the interim, the inlet dampers may have to be opened manually and the supply fans may have to be replaced because of damage from the fire in the control room. A new isolation switch will also be installed to operate the HV-8812B, RWST to RHR pump suction valve; meanwhile that suction valve must be operated manually. Containment spray pump train A is also tripped to prevent or stop its spurious operation. The train B spray pump was isolated during phase A when the 4160-B bus was stripped.

During phase C, which is completed within 20 minutes after the control room has been evacuated, operators trip the valve breakers and verify the position of and manually operate, if necessary, valves in the component cooling water (CCW) system to assure proper CCW system lineup, then start CCW pumps B and D. A new isolation switch will be installed to ensure that valve HV-70B closes; HV-70B is an air-operated solenoid-controlled CCW isolation valve for the radwaste building. In the interim, by pulling a fuse to kill dc power to the solenoid valve, the isolation valve will close.

During phase D, which is completed within 30 minutes after the control room has been evacuated, operators use charging pump B to line up the charging system and initiate RCP seal injection flow by using the RWST as a source. If the MSIVs were not closed before the control room was evacuated, they will now be closed using a portable 125-volt dc power source and wires will be cut to ensure the MSIVs remain closed. Also during phase D, operators ensure that the condensate storage tank (CST) is lined up to the turbine-driven AFW pump. At this time the operator at the ASP may use the turbine-driven pump in lieu of or in addition to the motor-driven B pump.

During phase E, which is completed 60 minutes after the control room has been evacuated, the operators will ensure the availability/operability of systems and components required for long-term hot standby. These include containment air cooling, fuel oil transfer system, and the isolation of minor potential blowdown paths such as the reactor head vents, steam generator blowdown system, excess letdown line, and the MSIV bypass valves. During phase E the charging system is lined up to charge through the boron injection tank (BIT) to allow boration and at Callaway the ESW system flow return is lined up to the cooling tower.

During phase E, operators pull identified fuses to prevent reactor head vent valves, excess letdown isolation valves, and the MSIV bypass valves from opening spuriously. This is acceptable since the valves are all normally closed, fail-closed valves and, except for the bypass valves, require multiple hot shorts to result in a blowdown path since there are two isolation valves in series. These are small blowdown paths (1-inch) and would result in a limited rate of release. Regarding the MSIV bypass valves, additional downstream valves would have to spuriously open in order to result in steam releases. Also, if instrumentation on the ASP indicates that these spurious operations had occurred, these steps could be taken any time before reaching phase E. Likewise, the steps to isolate the PORVs, atmospheric dump valves on steam generators A and C, or the steam generator blowdown system could be taken at any time if the instrumentation at the ASP indicated that isolation was necessary. These steps do not require pulling or replacing any fuses. Although it would take multiple hot shorts to cause spurious opening of the series RHR suction isolation valves, the breakers to one valve in each path will be tripped during normal operation to preclude a fire-induced loss-of-coolant accident (LOCA).

The final and long-term phase, phase F, includes (1) operations to assure the operability of the ESW system's self-cleaining strainers, (2) power and ventilation are established to the electrical equipment room for the cooling tower, and (3) the cooling tower fans are started. If necessary, the ESW system is lined up to the AFW system if the condensate storage tank is depleted.

Many of the manaul operations performed during phases A through F are precautionary to prevent spurious operations of valves and/or pumps. It is not expected that all spurious operations will occur and, in all likelihood, many of the manual valve lineups described in the procedures for the cooling water systems would only be valve lineup checks. Actual manipulation of a valve may be required only if the valve spuriously moved to an undesired position before isolating control power from the control room, or if the valve's normal position was not that desired for the post-fire lineup.

On the basis of the staff review of the phased procedural approach outlined with the August 23, 1984 submittal, and the interim procedures identified for use until the installation of the five new isolation switches and the modifications to four of the existing switches, the staff concludes that the SNUPPS alternative shutdown capability is acceptable pending the following conditions:

(1) Because of the time needed to design, procure, install and test the isolation switches, the staff has decided that the Callaway licensee does not have to install the isolation switches before a full-power license is issued. The basis for this deferral is staff judgment that the interim procedures provide a level of safety comparable to the design with the modified and new isolation switches for the time period of the first operating cycle.

(2) Before exceeding 5% of rated power, the licensee will revise his procedures for responding to a fire in the control room in accordance with the licensee's submittal of August 23, 1984 and will train operators to the revised procedures, including the interim procedures.

(3) In addition, the staff will condition the license to require the licensee to install the five new isolation switches and modify the four existing isolation switches that were identified in the August 23, 1984 submittal:

    (a) Before startup following the first extended outage of known duration (greater than two weeks) occurring after February 15, 1985, or

    (b) Before startup following the first refueling outage.

If the full-power license is not issued before March 1, 1985, the staff will require that the new isolation switches be installed and existing isolation switches be modified before exceeding 5% of rated power.

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13 CONDUCT OF OPERATIONS

13.1 Organizational Structure and Qualifications

13.1.2 Operating Organization

13.1.2.1 Operational Readiness

During a management meeting on May 9, 1984, with representatives of Union Electric Company (UE), the applicant was requested to provide the staff with an assessment of the readiness of UE to operate the Callaway Plant. By letter dated June 1, 1984, the licensee submitted a copy of the Operational Readiness Review. The review consists of an overall evaluation of the present status of UE relative to preparations for plant operation, and includes detailed information concerning the status of each of the major onsite organizational elements supporting the Callaway Plant as well as the technical support groups and the quality assurance groups that are part of the UE corporate organization. For each of these groups, the review presents a summary of present status relative to

(1) departmental procedures
(2) staffing
(3) personnel qualifications and training
(4) consultant utilization
(5) staff performance
(6) experience

The current staffing for operations at the Callaway Plant consists of approximately 520 persons out of an authorized total of 56?. In addition, the applicant employs about 100 consultant personnel to assist as necessary at strategic locations throughout the organization. There is a separate security force of about 200 persons, and the licensee currently has a maintenance contract that supplies about 200 craftsmen to supplement the UE activities. The experience of UE personnel assigned to the plant staff and the consultants who will remain at the plant beyond June 30, 1984, is shown in Table 13.1. This table shows that there are 176 individuals who hold degrees, primarily in engineering or a related science; the plant staff and consultants represent an accumulated total of nearly 1,470 years of nuclear Navy experience, 778 years of nonnuclear power plant experience, and about 2,175 years of nuclear power plant experience, of which about 275 years were accumulated at operating nuclear power plants. The consultants will remain at the plant until UE management is satisfied that the UE personnel are sufficiently experienced so that the consultants can be released.

When the experience of the consultants is taken into account, the number of personnel and the experience levels for Callaway compare favorably with the number and levels at the Washington Nuclear Project, Unit 2 (WNP-2) plant, which received a full-power license on April 13, 1984. The WNP-2 staff consisted of 406 people with total nuclear experience of 3,825 man-years, total boiling-water-reactor (BWR) experience of 1,549, man-years, and total operational BWR experience of 565 man-years.

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The plant has a full-time, dedicated training staff, which now consists of about 30 people and which has been augmented at times by additional personnel from Westinghouse and other contractors. A plant reference simulator is located on site and is used for training plant personnel. In total, plant staff personnel have received more than 360,000 hours of training.

The applicant has 42 successfully licensed personnel, of which 21 are senior licensed operators and 14 are licensed operators available for shift operation. Two management and five training department personnel also hold senior operator licenses. In addition, 10 management and engineeering personnel have been certified as senior operators but have not taken the NRC examination. The 42 licensed personnel now on staff represent a 97% success rate for the licensed operator training program. All of the licensed shift personnel have been assigned to operating plants similar to Callaway for from 4 to 6 weeks of observation/participation training.

However, only one of the senior licensed operators on shift has had at least 6 months of licensed experience at a hot, operating plant of the same type as Callaway. To compensate for this shortage of operational experience, the licensee has retained the services of operations advisors (OAs), who will provide this hot-operations experience to those shifts that do not have a shift member with such experience. To date, the licensee has certified four OAs to the NRC as being trained and qualified to provide advice to the operating shifts. This is enough to provide one such advisor or an experienced senior licensed operator on each of the five shifts that the applicant plans to use during the startup and test program. (Six shifts are planned during commercial operation of the plant.) During a meeting with the licensee on May 30, 1984, the staff was informed that the licensee also plans to have two additional OAs trained and qualified to provide backup capability if any of the four advisors now available leave.

The staff has evaluated the qualifications of the operations advisors, including previous experience, the training program which they underwent at Callaway, and the written and simulator/oral examinations that they took at the completion of the training, and concludes that the advisors are technically qualified to assume their roles on shift. The staff did, however, identify a concern in the OA program. This concern pertained to the specific duties of the advisors. The procedure that defined the role of the advisors, APA-ZZ-00010, provided only very general guidance on their duties. By letter dated July 16, 1984, the licensee provided a revised procedure that detailed the specific duties of the shift advisors. On the basis of its review of the information submitted in the July 16, 1984, letter, the staff has concluded that the licensee has adequately addressed the above concern. A detailed discussion of the OAs is provided later in this section.

The plant staff has developed more than 2,700 procedures (administrative, departmental, and surveillance). Although there still are a few procedures under development, none of these are necessary for fuel load, power-ascension testing, or plant operation. Writing of procedures is essentially complete. Many of the procedures have been used for control of plant activities during the hot functional testing and the preoperational program. The plant personnel thus have had an opportunity to use the procedures in practice and to revise them as necessary when problems were uncovered.

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To help ensure the safety of initial plant operations, the applicant has established a Senior Operations Advisory Panel (SOAP). This group, a subcommittee of the On-Site Review Committee, is composed of individuals having extensive operations management experience in commercial nuclear power plants. Each of the three panel members has had previous commercial nuclear operating experience, two in the area of operations management and the third in the area of operational quality assurance. The panel is now functioning to provide a continuing assessment and evaluation of the day-to-day operations at the Callaway Plant. It will pay particular attention to events that may be attributable to lack of qualification or experience of the plant staff. The SOAP has ready access to all levels of management up to and including the Vice President - Nuclear for the purpose of obtaining information, researching causes or solutions, and making recommendations on corrective or remedial action. The panel will focus on the onsite nuclear operations, but it also has the freedom to look into off-site nuclear support functions. Panel members will continue to perform their normal duties, but their primary function will be the panel activities until the plant has attained commercial operation or for 1 year, whichever is later. The Operational Readiness Review includes a charter for SOAP that describes its purpose and scope, panel membership, period and method of operation, duties and responsibilities, and how its activities are to be documented. The panel is not designed to produce additional paper trails regarding plant operations, but rather to devote its time to overseeing plant operations to detect potential trouble spots before they occur and to recommend appropriate corrective action. Nontheless, the panel will make short status reports at significant points during the startup program, and the licensee has orally committed that these reports will be made available to the staff. The staff has discussed documentation of SOAP recommendation with the licensee, and the licensee has agreed to do so. Further, the licensee has agreed to have SOAP perform a special review of plant activities to assess the plant's readiness to proceed beyond 5% power.

The staff's evaluation of the Operational Readiness Report is that the licensee has a well-staffed operations and technical support organization with a considerable depth of experience in Navy nuclear, commercial nonnuclear, and commercial nuclear power plants. Much of the commercial nuclear power hot-operating experience is furnished by consultants, but the licensee plans to retain these individuals until the UE employees are sufficiently experienced to operate the plant safely without outside assistance. Although the licensed operators have only limited actual hot-operating experience in a licensed capacity at similar nuclear plants, all of the operators have been able to spend at least 1 month in observation/participation training at other plants. The many experienced consultants located throughout the organization, including the experienced operations advisors provided to those shifts lacking in previous hot-operating experience, should compensate for any shortages of previous hot experience among the plant staff personnel. Many of these consultants have been at Callaway for significant periods and have fully integrated with the plant personnel. The Senior Operations Advisory Panel, now functioning, should be able to provide additional oversight of early plant operations so that any problems stemming from lack of personnel qualifications or experience will be readily detected and corrective actions can be taken.

Overall, the staff concludes that, from the standpoint of plant staffing and qualifications and the availability of procedures, the Callaway Plant is ready to operate. The weakness noted earlier regarding the lack of definition of the

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specific duties of the operations advisors has been corrected, and the advisors and shift crews will be trained regarding the advisor's duties before the plant exceeds 5% power.

By letter dated March 13, 1984, the licensee advised the staff that there were not enough experienced senior operators to fully staff the operating shifts and that operations advisors would be used to satisfy the hot-participation experience requirements. During a May 9, 1984, briefing for the NRC at the Callaway Plant, the licensee discussed shift staffing and qualifications of the operations advisors. Information presented during the briefing included

(1) the duties and authority of OAs and their working relationship with operating shift personnel

(2) the training program for OAs and the written and oral examinations administered to OAs

(3) the medical screening program for OAs

(4) the program for evaluating performances of OAs

The staff reviewed in detail information obtained during the May 9 briefing, and, on May 16, 1984, the licensee submitted copies of the résumés of the advisors, whom the licensee has designated OAs.

The staff has now completed its review of OA qualifications; the training program presented to the OAs, including the written and simulator/oral examinations administered at the end of the training program; the procedure used to define the duties of the OAs; and additional requirements for the advisors. The criteria used for the staff's review are those stated in SSER 3 plus experience gained during the review of advisor programs at the Diablo Canyon and Grand Gulf plants.

(1) Operations Advisor Qualifications

The staff finds that the advisor to the Plant Manager is well qualified. He holds a BS degree and has completed the course work toward an MBA. He has had 7 years of experience in the Navy nuclear program; more than 3 years of experience at the Farley Plant in positions as Training Supervisor, Technical Superintendent, and Operations Superintendent, holding a senior reactor operator's (SRO) license for the latter 13 months of this period; and more than 2 years of experience as the Nuclear Plant Manager at Crystal River, Unit 3. He has served as the advisor to the Plant Manager at Callaway since February 1982.

Two of the OAs amply meet the experience requirements as specified by the industry in the February 24, 1984, proposal to the Commission. The third OA has not previously held an SRO license, but has had 19 months of experience as a reactor operator (RO) at San Onofre, Unit 1. His nuclear experience includes 16 months as a nuclear plant control operator (nonlicensed) at Palo Verde, 9 months as a startup and test engineer at Callaway, and 6 months as a consultant to the Shoreham Operations Department staff. He has been at Callaway as a

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consultant to the plant operations department since September 1982. The breadth of his experience and the fact that he was verified as having an indepth knowledge of overall plant operations and as having demonstrated leadership and supervisory skills led the staff to conclude that he is adequately qualified to serve as an OA at Callaway.

The fourth OA has had indepth experience at the Zion station, serving as equipment attendant, licensed equipment operator, and licensed reactor operator. Of the total of nearly 7 years at Zion, he held an RO license for more than 2 years and was assigned as a licensed control room operator for 18 months. He also spent 5 months at Marble Hill as a shift control supervisor (nonlicensed) just before his assignment to Callaway. He was employed by Callaway as an operating supervisor and will be trained and licensed as an SRO at the first available opportunity, but will serve temporarily as an OA. Considering the similarity of the Zion units to Callaway, the staff considers this individual adequately qualified to serve as an OA at Callaway.

The licensee has informed the staff that two additional advisor candidates will be hired and will be trained. The additional advisors will provide relief and support to the current group of advisors and will be available after completion of training and evaluation by the Callaway staff. The staff was also informed that the new advisors will meet the minimum qualifications requirements.

(2) Operations Advisor Training Program

Between February 6 and April 13, 1984, the OA training program was conducted in two 3-week segments. The program contained the following elements:

(a) self-study

(b) reactor and plant systems lectures

(c) lectures in Technical Specifications, including seminars on limiting conditions for operations

(d) lectures and seminars on station normal, abnormal, emergency, and administrative procedures

(e) simulator exercises, which include normal, abnormal, and emergency operation

Approximately 60 hours were scheduled for self-study. Simulator training was included in both segments for a total of 50 hours. The remaining time consisted of formal lectures and seminars. The training modules for formal lectures and simulator exercises were drawn from the regular plant training program.

At the end of the training period, April 14, 1984, the OAs were evaluated by written and simulator examinations. The written examination was administered in three sections: Systems, Procedures, and Technical Specifications. The simulator examination consisted of an evaluation of the OA in the role of a supervisor during normal, abnormal, and emergency exercises and responses to oral questions during the course of the evaluation. The written and simulator examinations were witnessed by an examiner from Region III.

On completion of the written examination, the Callaway trainers and the Region III examiner independently graded the tests. The overall scores agreed within 5%. The Callaway trainers found that the scores of one of the OA's were marginal. That OA was given remedial assignments and later passed the makeup examinations. All OAs successfully passed the simulator/oral examinations. The Region III examiner concurred with these evaluations.

The staff concludes that the contents of the training program, including lesson plans, met the SER conditions and that the written examination was adequate to determine that the OAs had demonstrated proficiency in the subject matter. This is further supported by the Region III evaluation and the staff's review of the examination questions. The staff's evaluation revealed that about 50% of the questions were at the senior operator level. The staff concludes that the simulator/oral examinations adequately evaluated the OA in a role as supervisor but fell short in evaluating the OA as an advisor. This issue is discussed further in the following section.

(3) Operations Advisor Procedure

The qualifications and responsibilities of the OA are contained in Sections 4.1.8 and 4.2.8 of Callaway's Administrative Procedure APA-ZZ-00010. This procedure established the Operations Department's organizational structure and functions and also includes the responsibilities of all personnel in the Operations Department. Revision 2 of APA-ZZ-00010, which first defined the OA position, was developed on April 12, 1984, and issued on April 26, 1984. Training sessions for the shift crews regarding the role of the OA were conducted during the period May 1-8, 1984. However, because the examination of the OAs was conducted on April 14, 1984, and Revision 2 of APA-ZZ-00010 was not developed until April 12, 1984, it is the staff's opinion that the OAs were trained and evaluated without use of the revised procedure. Prior to exceeding 5% of rated power the staff will ensure that the shift crews are retrained on the revised procedure.

The OA responsibilities set forth in Section 4.2.8 of APA-ZZ-00010 include:

(a) The OA will advise the Operations Department on matters pertaining to the safe, legal, and efficient operation of the plant.

(b) The OA is assigned under the administrative direction of the Superintendent of Operations (SO).

(c) The Shift Supervisor (SS) will assign the OA responsibility a the senior operator level with commensurate authority.

(d) The assignments (by the SS) shall not include those that require an operator's license and do not include direction of licensed operators in the performance of duties.

(e) The OA may recommend appropriate actions (including shutdown) to the SS.

(f) The OA shall have direct access to the SO or the emergency duty officer to resolve any disagreements that may affect safe operation of the unit.

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The staff agreed with the limitations that restrict the advisor from performing or directing licensed activities. In addition, the staff concurs that the advisor recommend appropriate actions and resolve disagreements that may affect safe operation. However, the staff disagrees with the position that the advisor be assigned responsibility solely by the SS. The advisor's responsibilities should be specific and approved by the Callaway management.

The staff discussed this matter with the Plant Manager during a meeting in Bethesda, Maryland, on May 30, 1984. It was agreed during the meeting that the licensee would revise the procedure so that the duties of the OAs would be clearly stated and that, before 5% power is exceeded, both the advisors and the shift crews would be trained on this revised procedure. In a letter dated July 16, 1984, the licensee advised the staff that the shift advisors and crews would be retrained on the new procedure before 5% of rated power is exceeded. The staff has reviewed this information and finds it acceptable.

(4) Additional Advisor Requirements

The licensee plans to perform quarterly appraisals of the OA performance utilizing the standard evaluation used for all Callaway management employees. The staff concurs with this method of evaluation.

The OAs have been given physical examinations in compliance with applicable regulatory guides, NUREGs, sections of 10 CFR, and American National Standards Institute standards. The staff agrees with the standards; however, it has no knowledge of the results.

The licensee had not indicated if the OAs will participate in the licensed operator requalification program. The staff (and industry reviewers at other plants) has recommended that advisors be enrolled in the requalification training program and, when possible, attend training sessions with their assigned crews. The staff has discussed this matter with the licensee who has agreed that the OAs will participate in the Callaway licensed operator requalification training.

(5) Conclusions

The staff review of the OA qualifications indicates that the four OAs meet the requirements or have demonstrated equivalent experience. In addition, the licensee plans to train and qualify two additional OAs to provide relief and support to the current advisors.

The staff concludes that the Callaway training program for OAs prepared them to assume the technical role as advisors. This conclusion is supported by a review of the course outline, lesson plans, and simulator exercises as well as the written and simulator/oral examinations.

Overall, the staff concludes that the licensee has provided for adequate operating experience on shift to satisfy the current requirements for issuance of a full-power amendment; therefore, the 5% portion of LC 2.C.(8)(a) has been satisfied.

Table 13.1 Experience summary: Licensee personnel

Department	Non-nuclear power plant experience (months)	Nuclear plant experience before fuel loading (Callaway & others) (months)	Commercial nuclear experience post-fuel loading (months)	Navy nuclear experience (months)	No. with college degree
Planning & Scheduling	947	1,765	215	776	10
Compliance	182	3,027	228	1,224	23
Training	82	1,214	144	2,468	9
Maintenance	3,469	2,641	49	693	8
Administration - Records	-	259	15	0	3
Administration - Services	24	217	-	0	2
Health Physics	40	1,127	377	3,046	9
Radwaste	20	504	226	1,700	2
Chemistry	146	518	37	1,487	7
Instrument & Control	903	2,553	733	2,121	8
Engineering	881	3,530	254	785	55
Operations	1,520	2,237	493	2,549	16
Materials	265	1,638	11	252	13
Project Schedule	85	494	337	77	4
Management Staff	625	457	67	295	5
Security	142	624	108	156	2
Totals	9,331	22,805	3,294	17,629	176

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14 INITIAL TEST PROGRAM

The licensee had proposed a number of changes to the initial test program in Chapter 14 of the SNUPPS FSAR. These changes were submitted by letters dated May 15 and May 29, 1984. In all changes, the test objectives remain unchanged. In most of the changes proposed, the objective, test method, and acceptance criteria remain unchanged. In a few changes, the test method has been modified so that it is current with vendor-recommended methodology. These proposed changes have been grouped and discussed in the following paragraphs:

<u>Conformity With the "As-Designed and Built Plants"</u>

Preoperational Tests:

(1) S-03AE02, Steam Generator Level Control Test--This abstract is updated to reflect a previously implemented design change to the steam generator level control system. The SNUPPS plants now employ a constant 50% level setpoint, rather than the load-following design on which the original abstract was based.

(2) S-03GN02, Control Rod Drive Mechanism (CRDM) Cooling Preoperational Test--This abstract is changed to apply the acceptance criteria to only the "appropriate" CRDM fan breakers. Only two of four installed supply breakers are designed to open on receipt of a safety injection signal; these are the "appropriate" brakers.

(3) S-04HC03, Resin Transfer Preoperational Test--This abstract is revised to refer to only one chemical drain pump (vs. "pumps") in accord with the SNUPPS design.

Power Ascension Program:

(4) S-07BB04, Reactor Coolant System Flow Coastdown Test--This abstract is revised to delete reference to testing from "various operating configurations." All testing will be initiated from the four-loop operating configuration; the three-loop configuration has been deleted because a three-loop license will not be issued.

(5) S-07AB01, Steam Generator Level Control Testing--This abstract does not require more accurate calibration than that which will prevent spurious flow mismatch alarms due to design change single setpoint (see Change (1)) in this abstract.

(6) S-07SF04, Rod Position Indictor--The revisions in this abstract (a) modify the prerequisites from hot shutdown to cold shutdown condition, (b) change control rod bank withdrawal from 20-step increments to 24-step increments, and (c) verify the shutdown bank positions are only at 18 or less steps and at 210 or greater steps. Revision (a) applies to test flexibility, and because the revision does not compromise the test objective, it is acceptable. Revisions (b) and (c) reflect the as-designed system and are, therefore, acceptable.

The above six changes are made to reflect the as-built or as-licensed plant and, therefore, are acceptable.

Current Vendor-Recommended Methodology

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Power Ascension Test Program:

(7) S-070008, Power Coefficient Determination Test--This abstract is revised to reflect the power coefficient test methodology now recommended by Westinghouse; the superseded methodology had been in use at the time the FSAR was orginally submitted.

(8) S-07SF04, Rod Position Indication System Test--This abstract is revised to reflect the rod position indication system test methodology now recommended by Westinghouse.

(9) S-070018, Calibration of Steam and Feedwater Flow Instrumentation at Power Test--The acceptance criteria for this abstract are revised to account for decreased instrument accuracy at lower reactor power levels, consistent with Westinghouse recommendations.

Changes (7) through (9) do not change the intent of the test objectives for these tests. The changes in power coefficient measurement methodology are consistent with other approved test programs that use the vendor-recommended methodology and are, therefore, acceptable.

Change (8) expands the testing so that supplemental data recommended by the vendor can be gathered and are acceptable. Change (9) reduces the number of calibration points to those near full power (75 and 100%). The feedwater and steam flow instruments supply signals, in addition to individual readouts, to both a steam-feedwater flow mismatch alarm and trip signal. The steam-feedwater flow trip signal is required at full power to be coincident with an indication of low level in the steam generator for protection from loss of heat sink. Because the trip is primarily required at higher power levels, the greater calibration accuracy is necessary on the upper portion of the flow curves near the operating point. This is generally consistent with flow instrument calibration and will result in reasonable accuracy at lower flows as well. On the basis of these considerations, the staff finds that Change (9) is consistent with vendor recommendations and is acceptable.

Miscellaneous Administrative

The changes in this category are miscellaneous administrative ones because in most cases they represent changes to nonsafety-related tests that are primarily corrections of an administrative nature.

(10) S-03BB09, Reactor Coolant System Flow Measurement Test--This abstract is changed to apply the acceptance criteria to total reactor coolant system flow rate rather than individual loop flow, making the preoperational test consistent with the startup test. Both total flow and individual loop flows satisfied the acceptance criteria in the Callaway test.

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(11) S-04BL01, Reactor Makeup Water System Preoperational Test--This abstract was changed from nonsafety related to safety related in FSAR, Revision 13, because the test included response of the reactor makeup water system containment supply valve to a containment isolation signal (CIS). The change was not appropriate, however, and the abstract is being changed back to nonsafety related; the safety-related test of this CIS valve is performed separately (Abstract S-03SA01).

(12) S-04AC02, Turbine Trip Test--This nonsafety-related abstract is revised to correct a typographical error. A turbine trip signal is initiated on loss of electrohydraulic control 125-V dc power with turbine speed below 75% (not 25%).

Changes (11) and (12) make corrections to the abstracts and are acceptable. The administrative change to make the preoperational test (Change (10)) consistent with the startup test for reactor coolant system flow measurement is acceptable because the test is essentially unchanged. The loop flows are still measured by loop elbow differential pressure, converted to flow, and summed for total reactor flow.

The licensee has recommended modifying Test Abstract S-04HC01, "Solid Waste System Preoperational Test," to exclude the variable capacity positive displacement pumps from the acceptance criteria of this nonsafety-related test. The licensee has indicated that the positive displacement pumps do not produce a flow-head curve with which their performance can be compared. This reason is inadequate to justify deleting the positive-displacement pumps from the test program because there must be design specifications with which these pumps can be compared. However, because the pumps perform no safety-related function, the staff concludes that the change is unnecessary, but acceptable.

Unacceptable Change

This change pertains to Test Abstract S-07SE01, "Nuclear Instrumentation System." The change essentially modifies the test abstract so that only the testing on the source range monitor need be completed before fuel loading. Inadequate justification has been submitted to allow delay of intermediate-range testing beyond fuel loading. Although it is not necessary to have the power range monitors available for fuel loading, it has been a traditional safety practice to have both source-range monitors and intermediate-range monitors tested and functional for fuel loading as stated in Regulatory Guide 1.68, "Initial Test Programs for Water-Cooled Nuclear Power Plants," Appendix A, Test 2.g. The reason that the intermediate-range monitors should be functional during fuel loading is that if an inadvertent criticality were to occur, the source-range monitors could saturate. Thus, the intermediate-range monitors would be useful to provide a record of the power transient. Therefore, the modification deferring the power-range monitor testing is acceptable; but the change that would defer testing of the intermediate-range monitors until after fuel loading is not acceptable. The intermediate-source-range monitors were tested and operable before fuel loading.

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15 ACCIDENT ANALYSIS

15.4 Radiological Consequences of Design-Basis Accidents

15.4.4 Steam Generator Tube Rupture Accident

In the previous supplement to Callaway's SER (SSER 3), the staff indicated that, in order to satisfactorily resolve the steam generator tube rupture (SGTR) issue, additional information was required regarding the SGTR safety analysis, including the effects of loss of offsite power, confirmation of operator action times assumed, and the effects of steam generator overfill on secondary safety valve operability. Nevertheless, the staff concluded that there was sufficient assurance that the Callaway Plant could operate safely for one fuel cycle, before the SGTR issue is fully resolved, for the following reasons:

(1) All components necessary for mitigation of the design-basis SGTR are safety related.

(2) The Callaway Plant steam lines and supports are designed for the loads resulting if the steam lines are filled with water.

(3) There is a low probability of an SGTR, approaching the severity of the design-basis event, especially during the first cycle of operation. On the basis of the above conclusions, the staff conditioned the license to require satisfactory resolution of this issue before startup following the first refueling.

Subsequent to issuance of SSER 3, additional information has become available regarding operator actions and the associated times to mitigate the consequences of SGTR. On the basis of recent plant simulator runs and preliminary thermal hydraulic calculations performed by Westinghouse, operator action can be expected within a time frame compatible with mitigation of SGTR consequences. Thus, termination of primary to secondary leakage by pressure equalization can be expected within a time frame necessary to prevent steam generator overfill. The staff continues to believe that the consequences of an SGTR at Callaway can be adequately controlled by limiting the primary and secondary coolant system radioactivity concentrations by Technical Specification and by proper operator actions. The recent information regarding operator actions, delay times, and time to overfill indicates that sufficient time is available for proper operator actions to maintain the offsite radiological consequences below the staff's acceptance criteria. The staff further concludes that, subject to the receipt of the confirmatory information, the Callaway SGTR analysis is acceptable.

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22 TMI-2 REQUIREMENTS

22.2 Discussion and Conclusions

I.D.1 Control Room Design Review

Item I.D.1 of Task I.D, "Control Room Design," of the NRC Action Plan developed as a result of the accident at the Three Mile Island, Unit 2 (TMI-2) (NUREG-0660), states that operating licensees and applicants for operating licenses will be required to perform a Detailed Control Room Design Review (DCRDR) to identify and correct design discrepancies. The objective, as stated in NUREG-0660, is to improve the ability of control room operators in nuclear power plants to prevent or cope with accidents, if they occur, by improving the information provided to them. Supplement 1 to NUREG-0737, dated December 17, 1982, confirmed and clarified the DCRDR requirement in NUREG-0660. As a result of Supplement 1 to NUREG-0737, each applicant or licensee is required to conduct the DCRDR on a schedule negotiated with NRC.

NUREG-0700 describes four phases of the DCRDR to be performed by the applicant and licensee. The phases are:

(1) planning
(2) review
(3) assessment and implementation
(4) reporting

NUREG-0801, Draft, "Evaluation Criteria for Detailed Control Room Design Review," provides the necessary criteria for evaluating each phase.

As a requirement of Supplement 1 to NUREG-0737, applicants and licensees are required to submit a program plan that describes how the following elements of the DCRDR will be accomplished:

(1) establishment of a qualified multidisciplinary review team

(2) function and task analyses to identify control room operator tasks and information and control requirements during emergency operations

(3) a comparison of display and control requirements with a control room inventory

(4) a control room survey to identify deviations from accepted human factors principles

(5) assessment of human engineering discrepancies (HEDs) to determine which HEDs are significant and should be corrected

(6) selection of design improvements

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(7) verification that selected design improvements will provide the necessary correction

(8) verification that improvements will not introduce new HEDs

(9) coordination of control room improvements with changes from other programs such as safety parameter display system (SPDS), operator training, Regulatory Guide 1.97 instrumentation, and upgrade of emergency operating procedures

The NRC requires each applicant and licensee to submit a summary report at the end of the DCRDR. The report should describe the proposed control room changes and implementation schedules, and should provide justification for leaving safety significant HEDs uncorrected or partially corrected.

The NRC will evaluate the organization, process, and results of each DCRDR. The evaluation of the applicant's and licensee's DCRDR efforts will consist of the following, as described in NUREG-0801:

(1) an evaluation of the program plan report submitted by the licensee/applicant

(2) a visit to some of the plant sites to audit the progress of the DCRDR programs

(3) an evaluation of the licensee/applicant DCRDR summary report

(4) a possible preimplementation audit

(5) the preparation of an SER that will present the results of the NRC evaluation

Significant HEDs should be corrected. Improvements that can be accomplished with an enhancement program should be done promptly.

The Standard Nuclear Unit Power Plant System (SNUPPS) submitted the DCRDR Summary Report for Callaway Plant, Unit 1, on February 2, 1984. The staff conducted an onsite audit February 27 through 29, 1984, and transmitted an audit report to the licensee on June 5, 1984. The licensee developed responses to preliminary findings reported by the staff at the exit briefing of the onsite audit, and submitted these responses to the NRC on March 21, 1984. The licensee committed to submit an environmental survey report on the control room along with a revision to the DCRDR Summary Report before exceeding 5% power operation. This was made a condition of the operating license (Callaway License Condition C.(9)(a)). The licensee conducted the environmental survey in April 1984, and telephoned preliminary results to the staff on April 20, 1984. On June 29, 1984, the licensee submitted Revision 1 to the DCRDR Summary Report documenting the results of the environmental survey and resolutions to other items as proposed in the submittal of March 21, 1984. By letter dated June 29, 1984, the licensee proposed a modification to the auxiliary shutdown panel room to improve the operator's ability to read displays located very high on the panel. The staff has reviewed the above documentation of the Callaway DCRDR Program and provides the following summary of the degree to which the requirements of Supplement 1 to NUREG-0737 were satisfied:

Planning Phase

After reviewing the SNUPPS DCRDR Program Plan submitted in June 1983, the staff concluded that it was incomplete and did not address some of the elements in sufficient detail to establish how the element would be accomplished. After a meeting with the staff on October 25, 1983, SNUPPS submitted a revised plan on November 28, 1983. In addition, the DCRDR Summary Report contained samples of the forms used in documenting the methodologies and activities of the DCRDR.

The concerns expressed after the review of the original DCRDR Program Plan were as follows:

(1) qualifications of the human factors contractor and other engineering and training personnel

(2) involvement of the human factors consultant in the DCRDR

(3) level of involvement of each of the disciplines participating in the DCRDR for each DCRDR task

(4) organization of management for the DCRDR

With the exception of the level of involvement of an experienced human factors engineer in the System Function Review and Task Analysis (SFR&TA), subsequent discussions with SNUPPS and utility personnel and supplemental documentation satisfied these concerns. The SNUPPS DCRDR management structure and the qualifications and involvement of personnel were adequate to conduct a satisfactory DCRDR.

Review Phase

The activities included in SNUPPS's review phase are:

(1) operating experience review
(2) system function review and task analysis
(3) control room inventory
(4) control room survey
(5) verification of task performance capabilities
(6) validation of control room functions

Activities 2 through 5 address specific DCRDR requirements contained in NUREG-0737, Supplement 1.

(1) Operating Experience Review

SNUPPS recognizes the value of operating experience input in the DCRDR and although this is not a requirement of NUREG-0737, Supplement 1, they appear to have performed a review of operating experience which will provide valuable insights and feedback for other DCRDR activities.

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(2) System Function Review and Task Analysis

Besides the limited use of human factors engineering expertise in the task analysis effort, the staff has several concerns about the approach taken by SNUPPS to define the required design characteristics of controls and displays. The points below summarize these concerns:

(a) No analysis was conducted to define the required characteristics of "digital" (discrete) controls or displays.

(b) Because plant-specific documentation was used to identify some of the design requirements against which plant-specific instrumentation was compared, the verification of instrument suitability may not have been valid.

(c) On the basis of the SFR&TA writeup, examples of continuous monitoring and modulating control tasks, and the sample Task Analysis and Verification Worksheet, it is unclear what analysis, if any, was conducted to determine the information and control characteristics required by operators to accomplish their tasks.

(d) There appears to be inconsistency in the requirements specified for certain parameters in Appendices B and F of the Summary Report (F and J of Revision 1 to the Summary Report).

(3) Control Room Inventory

The inventory of controls and displays in the control room that is used in the DCRDR consists of plant design drawings and specifications. In itself, the inventory of controls and displays appears to be comprehensive and should have provided adequate support to the DCRDR as an information source.

(4) Control Room Survey

The control room survey work was initiated as the Preliminary Design Assessment (PDA) in 1980 using NUREG/CR-1580 as the source of evaluation criteria. After the issuance of NUREG-0700, SNUPPS performed a supplementary survey (SS) and a survey of the auxiliary shutdown panel (ASP). An environmental survey was performed in April 1984. The results of these surveys are summarized below.

(a) Preliminary Design Assessment

The open items from the NRC audit of the PDA which was performed in July 1981 have been determined to be adequately resolved and control room improvements implemented.

(b) Supplementary Survey

Appendix D of the DCRDR Summary Report (Appendix B of Revision 1 to the Summary Report) listed the human engineering discrepancies (HEDs) and SNUPPS's responses resulting from the SS. The audit team in the control room examined the HEDs from each of the nine sections of the SS. The resolutions to all findings in the SS were determined to be acceptable.

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(c) Auxiliary Shutdown Panel Review

Appendix E of the DCRDR Summary Report (Appendix C of Revision 1 to the Summary Report) lists the HED and SNUPPS responses resulting from the ASP review.

The audit team of the auxiliary shutdown panel examined the HEDs from the nine sections of the ASP review. Resolutions of all findings in the ASP review were finalized by two submittals from the licensee dated March 21, 1984, and June 29, 1984, and were determined acceptable as was the schedule for implementation of improvements.

(d) Environmental Survey

Results of the environmental survey indicate that air velocity in the control room is significantly higher than that recommended by human factors guidelines, ambient temperature is slightly higher than the recommended maximum temperature for personnel comfort, and the ambient noise level is at the maximum for unimpeded communications. The staff does not expect that any of these items individually will have an immediate, direct, detrimental effect on the safe operation of the Callaway Plant. However, the combination of discrepancies results in less than a desirable environment which would add unnecessarily to the overall stress level of the operator during emergency operations. In addition, if these undesired conditions exist when the systems and equipment are new, they cannot be expected to improve, but more likely will degrade with age. In general, the control room survey work (performed during the PDA, SS, and ASP reviews) and environmental survey activities were comprehensive and met the requirement of Supplement 1 to NUREG-0737 for "a control room survey to identify deviations from accepted human factors principles." In the context of this task, the staff finds that the SNUPPS review team is adequately resolving the HEDs identified and has improved the operability of the control room at Callaway.

(5) Verification of Task Performance Capabilities

The Callaway simulator is certainly an acceptable tool for verifying task performance capabilities. However, the staff is concerned that such verification (by performing tasks on the simulator) may lack objectivity through the natural tendency to uncritically accept, as suitable, that which already exists in the control room. Unless a set of predefined design requirements exists (from the task analysis), describing the characteristics of the controls and displays needed by the task, the only verification to be accomplished is that the controls and displays exist in the control room. Little can be said objectively about their suitability for performing the task. The acceptability of this task will be resolved as a part of the System Function Review and Task Analysis.

(6) Validation of Control Room Functions

SNUPPS performed two separate validations of control room functions. The first effort consisted of analyzing the video-taped walkthroughs of various procedures performed at the SNUPPS simulator at Zion. The findings from this analysis were incorporated as part of the PDA findings.

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The second effort consisted of analyzing the video-taped walkthroughs of the entire set of 41 Westinghouse Owners Group Emergency Response Guidelines (WOG ERGs) at the Callaway simulator. This validation effort appears to have been focused primarily on validating the WOG ERGs. In addition, SNUPPS took the opportunity of analyzing the video tapes to evaluate control room instrument and control consistency with the procedures, operator workload, and workstation flow or traffic. The six HEDs produced from this second validation effort reflect an adequate evaluation.

Assessment and Implementation Phase

(1) Assessment of HEDs

Although a prioritization process was carried out for the large majority of HEDs identified in the PDA and DCRDR, prioritization did not serve very often as a criterion for HED resolution or selection of design improvement. The SNUPPS approach was to correct as many HEDs as possible regardless of the assigned priority.

The staff finds that the requirement of Supplement 1 to NUREG-0737 regarding assessment of HEDs has been met.

(2) Selection of Design Improvements

The staff has reviewed all design improvements, both implemented and proposed, and finds that SNUPPS has met this NUREG-0737, Supplement 1, requirement.

(3) Schedules for Implementing HED Corrections

SNUPPS and utility personnel are responsive in accomplishing the improvements needed in the control room in an expeditious manner. Most improvements have already been completed. Only a few HEDs requiring long lead-time parts or more detailed design effort will be accomplished before startup from the first refueling outage.

(4) Verification That Improvements Will Provide the Necessary Corrections Without Introducing New HEDs

The procedure for this review includes (a) an evaluation of the redesign against the HED and recommended resolution, if provided, (b) a depiction and evaluation of significant changes on a full-scale mockup or control board drawing, and (c) performance of walkthroughs of selected procedures on either the full-scale mockup, the simulator, or the control room, after changes have been implemented. The revised design is scrutinized from a human engineering viewpoint by the DCRDR team and any feedback from the procedure walkthroughs is conveyed to the DCRDR team from utility operations and engineering personnel. The staff finds that this verification process satisfies the requirement of Supplement 1 to NUREG-0737.

(5) Coordination of the DCRDR With Other Improvement Programs

SNUPPS appears to be integrating the DCRDR with operator training, Regulatory Guide 1.97 instrumentation, development of emergency operating procedures (EOPs), and SPDS development in a manner that satisfies the requirement of Supplement 1 to NUREG-0737.

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Conclusion

By submitting Revision 1 to the DCRDR Summary Report on June 29, 1984, Callaway License Condition C.(9)(a) has been satisfied.

The staff concludes that the licensee, through the Standard Nuclear Unit Power Plant System, has conducted a DCRDR for Callaway Plant, Unit 1, that substantially meets the requirements of Supplement 1 to NUREG-0737 except in the area of System Function Review and Task Analysis and the subsequent activities resulting from that analysis. The staff requires additional information to determine the acceptability of the final control room design and the implementation of control room improvements. The licensee must conduct the task analysis to develop and document the following information:

(1) A description of how the design requirements were determined for the plant-specific documentation that was used to identify the design characteristics against which plant-specific instrumentation was compared.

(2) For each instrument and control used to implement the EOPs, an auditable record of how the needed instrument and control characteristics were determined. These characteristics should be derived through the task analysis process from the information and control needs identified in the background documentation of the ERG and from plant-specific information. Once these information and control characteristics have been developed, a review of the control room must be accomplished to verify the existence and suitability of the displays and controls to satisfy the information and control requirements. Should any discrepancies result, these must be analyzed to determine their safety significance, requirements for corrective action, and an implementation schedule. In a meeting with the licensee on July 13, 1984, an agreement was reached to accomplish the above effort by April 30, 1985. Completion and documentation of the task analysis should be made a condition of the operating license.

II.B.3 Post-Accident Sampling System

On the basis of its evaluation of the postaccident sampling system (PASS), the staff concluded in SSER 3 that 9 of the 11 criteria were acceptable. The following criteria remained unresolved:

Criterion (6) Provide a core damage estimate procedure to include radionuclide concentrations and other physical parameters as indicators of core damage.

Criterion (9) Provide information demonstrating applicability of procedures and instrumentation in the postaccident water chemistry and radiation environment, and retaining of operators on semiannual basis.

The licensee (by letter dated March 23, 1984) provided a procedure for estimating the degree of reactor core damage based on the Westinghouse Owners Group generic methodology, "Post-Accident Core Damage Assessment Methodology," Revision 1, dated March 1984.

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The procedure takes into consideration other physical parameters such as reactor core temperature data, reactor water level, sample location, and containment radiation levels and hydrogen concentrations. The staff has determined that these provisions meet Criterion (6) and are, therefore, acceptable.

The accuracy, range, and sensitivity of the PASS instruments and analytical procedures are consistent with the recommendations of Regulatory Guide 1.97, Revision 3, and the clarifications of NUREG-0737, Item II.B.3, "Post-Accident Sampling Capability, transmitted to the licensee on June 30, 1982. Therefore, they are adequate for describing the radiological and chemical status of the reactor coolant. The analytical methods and instrumentation were selected for their ability to operate in the postaccident sampling environment. The standard test matrix and radiation effect evaluation indicated no interference in the PASS analyses. The equipment and procedures used for the PASS will be tested or calibrated to maintain a high level of reliability. Training of operators will be conducted in accordance with the plant qualification program in conjunction with participation in semiannual emergency planning drills. The staff has determined that these provisions meet Criterion (9) of Item II.B.3 in NUREG-0737, and are, therefore, acceptable.

Conclusion

On the basis of its evaluation, the staff concludes that the postaccident sampling system now meets all 11 criteria of Item II.B.3 in NUREG-0737 and is, therefore, acceptable.

Because of the above review and because the postaccident sampling system at Callaway is operable, the staff finds that the licensee has satisfied License Condition 2.C.(9)(b).

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

The SNUPPS design incorporates a Class 1E microprocessor-based plasma display system to provide information of inadequate core cooling (ICC) to the control room operator. SSER 3 (SER Section 22, TMI Item II.F.2) provides a detailed description of this display system. The subject SSER required that the licensee submit the results of the qualification testing associated with the isolation devices used for the interface between the safety-related and nonsafety-related circuits. In response to this requirement, the licensee submitted information (Westinghouse Topical Report WCAP-10621, "Westinghouse Thermocouple/Core Cooling Monitor System Isolation Tests," dated July 1984) by letter dated July 26, 1984. The staff evaluation of this information follows.

The Class 1E microprocessor-based ICC monitoring system communicates (via isolation devices) with the Technical Support Center (TSC) and, optionally, with the plant computer. Circuitry associated with the TSC and the plant computer is nonsafety related. The purpose of the qualification testing was to demonstrate that the isolation devices used will provide adequate protection for the Class 1E portion of the design.

The test configuration allowed the Class 1E portion of the system to be monitored and evaluated while subjecting the nonsafety-related portion of the system to credible faults. Simulated thermocouple readings were fed to the thermocouple/core cooling monitor (TC/CCM) microprocessor. To simulate the normal

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configuration, the processed signals were then input to the Class 1E CCM, remote display, remote printer, and isolation device. These signals were monitored before and during the fault application (i.e., the remote visual display and printer were checked for any changes from pretest conditions). Also, an oscilloscope was connected to the input of the optical isolator so that any feedback (from nonsafety to safety) through the isolator could be detected while applying the faults. The test consisted of applying maximum credible faults (580 V ac, 120 V ac, and ±250 V dc), in the transverse mode, to the output side of the isolator.

The test results showed that the Class 1E input to the isolator was unaffected by the fault applications (i.e., the fault did not propagate through the optical isolation device). The Class 1E remote display and printer showed no changes from pretest conditions while the faults were being applied. Also, no spurious signals were noted on the oscilloscope which was connected to the optical isolator input.

Conclusion

The above test results confirm that the Class 1E microprocessor-based plasma display system will provide normal information to the operator while being subjected to a maximum credible fault and, thus, the acceptance criteria have been adequately met. The staff, therefore, concludes that the isolation capability of the optical isolator has been satisfactorily demonstrated through testing and that LC 2.C.(9)(c) has been satisfied. Thus, the isolator is adequate for use in the TC/CCM system.

APPENDIX A DRAFT

CONTINUATION OF CHRONOLOGY OF
NRC STAFF RADIOLOGICAL REVIEW OF CALLAWAY PLANT

March 16, 1984	Letter from licensee committing to comply with staff request concerning amending FSAR Section 3.9.3.2 and addressing staff requirement concerning full qualification of all safety-related equipment.
March 24, 1984	Letter from licensee committing to comply with staff request concerning amending FSAR Section 3.9.3.2.
April 27, 1984	Representatives from NRC, Union Electric Co., and SNUPPS meet in Bethesda, Md., to discuss the appeal of the containment structural integrity requirement given in the Callaway Technical Specifications. (Summary issued May 11, 1984)
May 1, 1984	Letter to licensee concerning offsite dose calculation manual (ODCM).
May 2, 1984	Letter from SNUPPS concerning revision in diesel generator start time.
May 3, 1984	Letter to licensee concerning Callaway Technical Specifications.
May 4, 1984	Letter to licensee requesting additional information.
May 5, 1984	Letter from licensee concerning financial qualification information.
May 7, 1984	Letter to licensee requesting additional information on Callaway Technical Specifications.
May 8, 1984	Letter from licensee concerning secondary water chemistry monitoring and control program.
May 9, 1984	Representatives from NRC and Union Electric Co. meet in Fulton, Mo., for a site visit before licensing of the Callaway plant. (Summary issued June 5, 1984)
May 9, 1984	Letter to Westinghouse withholding from public disclosure the SNUPPS/Westinghouse Interconnecting Wiring Diagrams and Process Control Block Diagrams associated with Pressurizer Pressure Input (CAW-84-25), Wolf Creek and Callaway.
May 11, 1984	Letter from licensee concerning Callaway Technical Specifications.

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May 14, 1984	Letter to licensee concerning Callaway Plant Technical Specifications - Appeals.
May 14, 1984	Letter to licensee transmitting a request for additional information on Technical Specifications.
May 15, 1984	Letter to licensee requesting additional information on preoperational testing.
May 15, 1984	Letter from SNUPPS concerning SNUPPS FSAR Chapter 14 changes applicable to Callaway.
May 15, 1984	Letter from licensee concerning control room design review.
May 15, 1984	Letter from SNUPPS concerning SNUPPS Technical Specifications Reactor Systems Branch issues.
May 16, 1984	Letter to licensee requesting additional information on Instrumentation and Control Technical Specifications.
May 16, 1984	Letter from SNUPPS concerning response to NRC review of SNUPPS FSAR Chapter 14 for Callaway license.
May 18, 1984	Letter to licensee transmitting 20 printed copies of NUREG-0830, Supplement 3, to the Safety Evaluation Report.
May 18, 1984	Letter from SNUPPS concerning Instrumentation and Control Systems Branch Technical Specification Questions.
May 21, 1984	Letter from licensee concerning implementation of Generic Letter 83-28.
May 21, 1984	Letter from licensee concerning Operating License Appendix B, Environmental Protection Plan, Non-Radiological.
May 23, 1984	Letter from SNUPPS concerning Callaway Plant Ultimate Heat Sink Technical Specification Requirements.
May 24, 1984	Letter from SNUPPS concerning pump and valve operability FSAR revision.
May 24, 1984	Letter to licensee concerning fire protection deferrals.
May 25, 1984	Letter from SNUPPS concerning SNUPPS Technical Specifications Reactor Systems Branch Issues.
May 25, 1984	Letter to licensee concerning operating shift staffing for Callaway Plant.
May 26, 1984	Letter from licensee concerning incorporation of SNUPPS document into application.
May 29, 1984	Letter from licensee concerning FSAR Chapter 14 abstract changes.

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May 29, 1984	Letter from licensee concerning Callaway Pump and Valve Inservice Testing Program.
May 30, 1984	Letter from licensee concerning vital area door controls.
May 30, 1984	Representatives from NRC and Union Electric Co. meet in Bethesda, Md., to discuss operational readiness.
May 31, 1984	Letter from licensee concerning Callaway Technical Specifications.
May 31, 1984	Letter from SNUPPS concerning SNUPPS Technical Specifications Reactor Systems Branch Issues.
June 1, 1984	Letter from licensee concerning operational readiness, Callaway Plant.
June 3, 1984	Letter from licensee concerning Callaway preservice examinations.
June 4, 1984	Letter from licensee concerning readiness for fuel load.
June 5, 1984	Letter to licensee concerning results of preimplementation audit of Callaway and Wolf Creek control room.
June 7, 1984	Letter to licensee concerning response to Generic Letter 83-28.
June 11, 1984	Letter to licensee concerning human factors discrepancies for the Callaway auxiliary shutdown panel.
June 11, 1984	Letter to licensee transmitting the Facility Operating License NPF-25 for the Callaway Plant, Unit 1. The license allows 5% of power operation (170 MWt). Enclosures include license with Technical Specifications A and B, Federal Register Notice, Amendment 1 to Indemnity Agreement No. B-93. Assessment of the Effect of License Duration on Matters Discussed in the FES.
June 11, 1984	Letter from licensee concerning Callaway Technical Specifications.
June 12, 1984	Letter to licensee requesting additional information on the steam generator tube rupture event.
June 13, 1984	Letter from SNUPPS concerning Callaway preservice inspection program: Supplemental data.
June 13, 1984	Letter from SNUPPS concerning Callaway preservice inspection program plan.
June 15, 1984	Letter to licensee concerning additional information needed to resolve 5% license conditions.

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June 21, 1984	Letter to licensee concerning preservice inspection program changes.
June 21, 1984	Letter from licensee transmitting two copies of Amendment 1 to Indemnity Agreement B-93.
June 21, 1984	Letter from SNUPPS concerning NUREG-0737, Item II.B.3, Post-Accident Sampling Capability.
June 26, 1984	Letter from SNUPPS concerning Revision 15 to SNUPPS FSAR.
June 26, 1984	Letter to licensee requesting additional information - conformance to Regulatory Guide 1.97.
June 27, 1984	Letter from SNUPPS concerning Callaway preservice inspection program.
June 29, 1984	Letter from licensee addressing staff requirement concerning full qualification of all safety-related equipment.
June 29, 1984	Letter from SNUPPS concerning Revision 1 to Detailed Control Room Design Review Program Summary Report.
June 29, 1984	Letter from licensee concerning control room design review - auxiliary shutdown panel.
June 29, 1984	Letter from licensee concerning Revision 8 to the Callaway Plant FSAR Site Addendum.
June 29, 1984	Letter from SNUPPS concerning inadequate core cooling instrumentation testing.
July 6, 1984	Letter from licensee transmitting an application for Amendment to Facility Operating License No. NPF-25, Revision to Technical Specification Figure 6.2-1.
July 6, 1984	Letter from licensee concerning incorporation of SNUPPS documents into application.
July 13, 1984	Representatives from NRC, SNUPPS, licensee, and Kansas Gas and Electric Co. meet in Bethesda, Md., to discuss the detailed control room design review for the SNUPPS plants. (Summary issued July 18, 1984)
July 16, 1984	Letter from SNUPPS concerning operating staff experience requirements.
July 18, 1984	Letter to licensee concerning review of design for automatic shunt trip for scram breakers.
July .6, 1984	Letter from SNUPPS transmitting WCAP-10621.

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July 26, 1984	Letter from licensee requesting an exemption from cold rod drop testing.
July 26, 1984	Letter to licensee requesting additional information on seismic and dynamic qualification.
July 27, 1984	Letter from SNUPPS concerning seismic and dynamic qualification.
July 31, 1984	Letter from licensee transmitting an application for partial exemption from Appendix J.
August 1, 1984	Letter from licensee transmitting a revision to Technical Specification Table 3.3-1.
August 7, 1984	Letter to licensee replying to Union Electric's request for deletion of cold rod drop testing.
August 8, 1984	Letter from licensee concerning number of shift rotations for Callaway plant.
August 10, 1984	Representatives from NRC, UE, and KG&E meet in Bethesda, Md., to discuss the isolation features during a control room fire at SNUPPS plants. (Summary issued August 10, 1984)
August 10, 1984	Letter from SNUPPS concerning fire protection review.
August 14, 1984	Letter from licensee requesting an extension of time for submittal of response to Generic Letter 84-15.
August 14, 1984	Representatives from NRC, UE, and KG&E meet in Bethesda, Md., to appeal the staff's position on the SNUPPS Safe Shutdown Analysis. (Summary issued August 17, 1984)
August 15, 1984	Representatives from NRC, UE, and KG&E, SNUPPS, and Bechtel meet in Bethesda, Md., to discuss the SNUPPS fire protection plan. (Summary issued August 22, 1984)
August 16, 1984	Letter from SNUPPS concerning conformance to Regulatory Guide 1.97.
August 22, 1984	Representatives from NRC, UE, KG&E, and SNUPPS meet in Bethesda, Md., to discuss the SNUPPS fire protection review. (Summary issued August 31, 1984)
August 30, 1984	Letter to licensee transmitting the Federal Register Monthly Notice - Applications and Amendments to Operating Licenses Involving No Significant Hazards Considerations - Callaway Plant.

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APPENDIX I

PRESERVICE INSPECTION RELIEF REQUEST EVALUATION

I. INTRODUCTION

This section was prepared with technical assistance of DOE contractors from the Idaho National Engineering Laboratory.

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

In letters dated January 18, February 7, February 13, February 24, March 26, April 9, June 13, and June 27, 1984, the licensee submitted requests for relief from ASME Section XI Code requirements which the licensee has determined to be not practical and provided supporting information pursuant to 10 CFR 50.55a(a)(2)(i). Therefore, the staff evaluation consisted of reviewing the licensee's submittals to the requirements of the applicable Code edition and addenda and determining if relief from the Code requirements was justified.

II. TECHNICAL REVIEW CONSIDERATIONS

A. The construction permit for the Callaway Nuclear Power Plant was issued on April 16, 1976. In accordance with 10 CFR 50.55a(g)(3), components (including supports), which are classified as ASME Code Class 1 and 2, have been designed and provided with access to enable the performance of required preservice examinations set forth in the 1977 edition of ASME Section XI, including the addenda through Summer 1978.

B. 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 examination and testing requirements which by themselves provide the necessary assurance that the pressure boundary components are capable of performing safely under all operating conditions reviewed in the FSAR and described in the plant design specification. As a part of these examinations, all of the primary pressure boundary full penetration welds were volumetrically examined (radiographed) and the system was subjected to hydrostatic pressure tests.

C. The intent of a preservice examination is to establish a reference or baseline before the initial operation of the facility. The results of subsequent inservice examination 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 flaws must be treated as new flaws and evaluated accordingly. Section XI of the ASME Code contains acceptance standards which may be used as the basis for evaluating the acceptability of such flaws.

D. Other benefits of the preservice examination include providing redundant or alternative volumetric examination of the primary pressure boundary using a test method different from that employed during the component fabrication. Successful performance of preservice examination also demonstrates that the welds so examined are capable of subsequent inservice examination using a similar test method.

In the case of Callaway Nuclear Power Plant, a large portion of the preservice examination required by the ASME Code was performed. Failure to perform a 100% preservice examination of the welds identified below will not significantly affect the assurance of the initial structural integrity.

E. In some instances where the required preservice examinations were not performed to the full extent specified by the applicable ASME Code, the staff may require that these examinations or supplemental examinations be conducted as a part of the inservice inspection program. Requiring supplemental examinations to be performed at this time would result in hardships or unusual difficulties without a compensating increase in the level of quality or 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 ASME Code, Section III, fabrication examinations.

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

Several of the preservice inspection relief requests involve limitations to the examination of the required volume of a specific weld. The inservice inspection (ISI) program is based on the examination of a representative sample of welds to detect generic degradation. In the event that the welds identified in the PSI relief requests are required to be examined again, the possibility of augmented inservice inspection will be evaluated during review of the licensee's initial 10-year ISI program. An augmented program may include increasing the extent and/or frequency of inspection of accessible welds.

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III. EVALUATION OF RELIEF REQUESTS

The licensee requested relief from specific preservice inspection requirements in submittals dated January 18, February 7, February 13, February 24, March 26, April 9, June 13, and June 27, 1984. On the basis of the information submitted by the licensee and review of the design, geometry, and materials of construction of the components, certain preservice requirements of the ASME Boiler and Pressure Vessel Code, Section XI, have been determined to be impractical. Imposing these requirements would result in hardships or unusual difficulties without a compensating increase in the level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(2), conclusions that these preservice requirements are impractical are justified as follows. Unless otherwise stated, references to the Code refer to the ASME Code, Section XI, 1977 edition, including addenda through Summer 1978.

A. Reactor Coolant Pump Seal Water Injection Line Welds, Category C-F, (16 welds with pipe diameter of 1.5 inches or less)

Component Identification (ID)	Component Weld Description
	Pump A Seal Water Injection Line Welds
2-BG-09-FW387	2" x 1 1/2" Reducer to 1 1/2" Pipe
2-BG-09-FW386	1 1/2" Pipe to Valve
2-BG-09-FW385	Valve to 1 1/2" Pipe
2-BG-09-FW384	1 1/2" Pipe to 2" x 1 1/2" Reducer
	Pump B Seal Water Injection Line Welds
2-BG-09-FW432	2" x 1 1/2" Reducer to 1 1/2" Pipe
2-BG-09-FW431	1 1/2" Pipe to Valve
2-BG-09-FW430	Valve to 1 1/2" Pipe
2-BG-09-FW429	1 1/2" Pipe to 2" x 1 1/2" Reducer
	Pump C Seal Water Injection Line Welds
2-BG-09-FW417	2" x 1 1/2" Reducer to 1 1/2" Pipe
2-BG-09-FW416	1 1/2" Pipe to Valve
2-BG-09-FW415	Valve to 1 1/2" Pipe
2-BG-09-FW414	1 1/2" Pipe to 2" x 1 1/2" Reducer
	Pump D Seal Water Injection Line Welds
2-BG-09-FW402	2" x 1 1/2" Reducer to 1 1/2" Pipe
2-BG-09-FW401	1 1/2" Pipe to Valve
2-BG-09-FW400	Valve to 1 1/2" Pipe
2-BG-09-FW399	1 1/2" Pipe to 2" x 1 1/2" Reducer

Code Requirements: Although the ASME Code Section XI does not require a volumetric examination of these welds, the licensee committed to perform augmented volumetric examinations.

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Relief Request: Relief is requested from performing the augmented volumetric examination on the subject welds.

Reason for Request: These 16 small-diameter (1.5 inches or less) pipe-to-component welds (4 welds each loop) could not receive a meaningful augmented volumetric examination because of a combination of the small pipe diameter and the minimum wall thickness. The applicant stated that these welds received an alternative liquid penetrant surface examination.

Staff Evaluation: This relief request is acceptable for PSI based on following considerations:

1. During fabrication ASME Code Section III requires the subject welds to receive a radiographic examination of the entire weld volume plus a surface examination.

2. For PSI an alternative liquid penetrant surface examination was performed.

3. The required ASME Section III examinations along with the supplemental liquid penetrant examination for PSI demonstrate an acceptable level of preservice structural integrity.

B. Class 1 Branch Pipe Connection Welds, Examination Category B-J (18 welds total)

Westinghouse Weld #	Callaway Weld #	Westinghouse Weld #	Callaway Weld #
	Loop 1		Loop 3
15	2BB-01-S102-3	15	2BB-01-S302-3
17	2BB-01-S105-5	17	2BB-01-S305-5
19	2BB-01-S101-5	18	2BB-01-S301-4
21	2BB-01-S101-8	20	2BB-01-S301-5
22	2BB-01-S101-9		
			Loop 4
	Loop 2		
		15	2BB-01-S402-3
15	2BB-01-S202-3	16	2BB-01-S402-4
17	2BB-01-S205-4	18	2BB-01-S405-5
19	2BB-01-S201-5	20	2BB-01-S401-5
21	2BB-01-S201-8	22	2BB-01-S401-6

Code Requirements: Table IWB-2500-1, Examination Category B-J, Item B9.31 requires a surface and volumetric examination for branch connection piping welds 2-inch nominal pipe size and greater.

Code Relief Request: Relief is requested from performing the required volumetric examination on the subject welds.

Reason for Request: Because of the materials of construction and the design and fabrication geometry of these corner type branch connections, the licensee has concluded that meaningful examination by ultrasonic

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methods is not feasible and that no other practical volumetric method is available. As an alternative, VT-2 examinations for leakage will be conducted in accordance with IWA-5240 during the system leakage and hydrostatic pressure tests.

Staff Evaluation:

This relief request is acceptable for PSI based on the following considerations:

1. During fabrication the subject welds have received liquid penetrant examinations and radiographic examination of the entire weld volume in accordance with ASME Code Section III requirements.

2. For PSI an alternative VT-2 examination for leakage was conducted during the system hydrostatic test and these welds have received the required surface examination.

3. The combination of required surface examination, visual examination for leakage and the Code required fabrication examinations demonstrate an acceptable level of preservice structural integrity.

C. Class 1 Examination Category B-J and Class 2 Examination Category C-F, Pressure Retaining Welds in Piping (23 welds total)

Code Requirements: Examination Category B-J requires a surface and volumetric examination of all pipe welds 4 inches nominal pipe size and greater. A surface examination only is required for pipe welds less than 4 inches nominal pipe size.

Examination Category C-F requires a surface and volumetric examination of all pipe welds over 1/2-inch nominal wall thickness. A surface examination only is required for pipe welds with 1/2 inch or less nominal wall thickness.

Code Relief Request: Relief is requested from performing 100% of the Code-required volumetric examination on each of the subject welds.

Reason for Request: The design of Class 1 and Class 2 piping systems has welded joints, such as, pipe-to-fitting and pipe-to-component, which physically obstruct all or part of the required Section XI examinations from the fitting or component side of the weld specified. The licensee has identified the piping system welds with geometric obstructions, identified the obstruction, and estimated the percent loss of volume coverage in the following table:

Component ID	Category	Description	Basis for Relief
System: Reactor Coolant			
2-BB-04-F015 2-BB-04-F014	B-J	4" Pipe to Valve	Valve geometry obstructs scan path with subsequent 5% loss of volume coverage.

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Component ID	Category	Description	Basis for Relief
2-BB-04-S011C	B-J	4" Pipe to 6" x 4" Reducer	Reducer geometry obstructs scan path with subsequent 15% loss of volume coverage.
2-BB-04-F003 2-BB-04-S02J	B-J B-J	6" Tee to 6" Pipe 6" Pipe to 6" Tee	Tee geometry obstructs scan path. 9% loss of volume coverage.

System: Accumulator Safety Injection

Component ID	Category	Description	Basis for Relief
2-EP-01-F-012	B-J	6" Pipe to 6" x 10" x 10" Tee	Tee geometry obstructs scan path. 9% loss of volume coverage.
2-EP-01-S003E	B-J	10" Pipe to 6" x 10" x 10" Tee	Tee geometry obstructs scan path. 20% loss of volume coverage.
2-EP-01-S003F	B-J	10" Pipe to 6" x 10" x 10" Tee	Tee geometry obstructs scan path. 20% loss of volume coverage.
2-EP-01-F002	C-F	10" Pipe to Valve	Valve geometry obstructs scan path. 13% loss of volume coverage.
2-EP-01-F016	C-F	10" Pipe to Valve	Valve geometry obstructs scan path. 13% loss of volume coverage.
2-EP-02-F020	B-J	6" Pipe to 10" x 10 x 6" Tee	Tee geometry obstructs scan path. 9% loss of volume coverage.
2-EP-02-F003	C-F	Valve to Pipe	Valve geometry obstructs scan path. 20% loss of volume coverage.
2-EP-02-F009	B-J	6" Pipe to 10" x 10" x 6" Tee	Tee geometry obstructs scan path. 9% loss of volume coverage.
2-EP-01-S013K	B-J	6" Pipe to 10" x 10" x 6" Tee	Tee geometry obstructs scan path. 9% loss of volume coverage.
2-EP-02-S003-6	B-J	10" x 10" x 6" Tee to 10" Pipe	Tee geometry obstructs scan path. 2% loss of volume coverage.
2-EP-02-S003-F	B-J	10" x 10" x 6" Tee to 10" Pipe	Tee geometry obstructs scan path. 2% loss of volume coverage.

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Component ID	Category	Description	Basis for Relief
System: High Pressure Coolant Injection			
2-EM-05-F007	B-J	6" Pipe to 6" Sweepolet	Sweepolet geometery obstructs scan path. 9% loss of volume coverage.
2-EM-03-F016	B-J	6" Pipe to 6" Nozzle	Nozzle geometry obstructing scan path. 13% loss of volume coverage.
2-EM-03-F014	B-J	Valve to 6" Pipe	Valve geometry obstructing scan path. 13% loss of volume coverage.
2-EM-03-F015	B-J	6" Pipe to Valve	Valve geometry obstructing scan path. 13% loss of volume coverage.
2-EJ-04-F017	B-J	6" Pipe to Valve	Valve geometry obstructing scan path. 13% loss of volume coverage.
System: Residual Heat Removal			
2-EJ-04-F026	B-J	12" Pipe to 12" Nozzle	Nozzle geometry obstructing scan path. 13% loss of volume coverage.
2-EJ-04-F031	B-J	12" Pipe to Valve	Valve geometry obstructs scan path. 13% loss of volume coverage.

Staff Evaluation: The staff has determined that the volumetric examination of the subject welds to the extent required by the Code is impractical because of the design of the piping systems. The licensee has conducted the preservice surface examinations on these welds. The staff therefore concludes that the limited Section XI ultrasonic examinations, the volumetric examinations performed during fabrication, and the hydrostatic test demonstrate an acceptable level of preservice structural integrity.

D. Reactor Pressure Vessel Welds, Category B-A and B-D (12 welds)

Code Requirement: Table IWB-2500-1, Examination Category B-A and B-D, require a 100% volumetric examination of the subject welds.

Code Relief Request: Relief is requested from performing 100% of the Code required volumetric examination.

Reason for Request:

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Weld ID	% Not Examined	Basis for Relief
Closure Head Weld 2-CH-103-101	35%	Control rod drive mechanism penetrations, three lifting lugs welded directly onto 2-CH-103-101, and obstructing closure head shrouding, preclude complete volumetric examination. Removal of closure head shroud during ISI will require 500 man-hours of effort, presenting considerable ALARA concerns and still would not permit complete weld coverage.
Lower Head to Shell Weld 2-RV-101-141	15%	When performing the perpendicular scan of the weld, the search unit cannot reach the weld area below the core support lugs because of the obstruction created when the examination head contacts the outside edge of each lug. A slight loss of coverage, reflected in the 15% total loss figure, is also encountered because of the lug obstruction when performing the parallel scan.
Lower Head to Dollar 2-RV-102-ISI	10%	Obstructions presented by the instrumentation nozzles when scanning the lower head to dollar plate weld and meridional welds preclude complete volumetric coverage.
Lower Head Meridional 2-RV-101-154A 2-RV-101-154B 2-RV-101-154C 2-RV-101-154D	10% (Combined)	
Flange to Vessel 2-RV-101-121	25%	Parallel scan portion of examination can only be done from lower side because of presence of flange taper above the weld. Complete perpendicular scan was done from flange mating surface.
Outlet Nozzles to Vessel 2-RV-107-121-A 2-RV-107-121-B 2-RV-107-121-C 2-RV-107-121-D	10% (each)	Approximately 10% of the total weld volume for each outlet nozzle is obstructed by contact between the examination head and the nozzle knuckle extending from the nozzle opening through the plane of the reactor pressure vessel inner diameter.

Staff Evaluation: The subject welds are partially inaccessible for examination because of the existing design. The staff concludes that the limited Section XI volumetric examination, the volumetric and surface examination performed during fabrication, and the hydrostatic test demonstrate an acceptable level of preservice structural integrity.

E. Pressurizer Dissimilar Metal Welds, Examination Category B-F (6 welds)

Code Requirement: Table IWB-2500-1, Examination Category B-F, requires a 100% volumetric and surface examination of the subject welds.

Code Relief Request: Relief is requested from performing 100% of the Code required volumetric examination.

Reason for Request:

Weld ID	Description	% Not Examined
2-TBB03-4-W	Relief Nozzle to Safe-end	20
2-TBB03-3-A-W	Safety Nozzle to Safe-end	20
2-TBB03-3-B-W	Safety Nozzle to Safe-end	20
2-TBB03-3-C-W	Safety Nozzle to Safe-end	20
2-TBB03-1-W	Surge Nozzle to Safe-end	15
2-TBB03-2-W	Spray Nozzle to Safe-end	5

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All examination limitations were caused by a combination of weld geometry and metallurgical obstruction from Inconel buttering used in the components.

Staff Evaluation: The subject welds are partially inaccessible as stated by the licensee. The staff concludes that the limited Section XI volumetric examination, the volumetric and surface examination performed during fabrication, and the hydrostatic test demonstrate an acceptable level of preservice structural integrity.

F. Fuel Pool Cooling and Cleanup Pipe Supports, Examination Category D-C (4 Supports)

Support ID

2-EC-04-R026
2-EC-04-R027
2-EC-04-R029
2-EC-04-R030

Code Requirement: Table IWD-2500-1, Examination Category D-C, Item D.3.2, requires component supports and restraints within the boundary of the above systems, for components exceeding 4-inch nominal pipe size, to receive a visual examination (VT-3) during each inspection period.

Relief Request: Relief is requested from performing the required preservice VT-3 examination.

Reason for Request: These pipe supports will be submerged in the spent fuel pool during the life of the plant. The supports contain partial penetration weldments. The subject weldments were visually inspected during construction; weld material controls, materials traceability and support configurations were also verified by the Callaway constructor during field fabrication.

Staff Evaluation: This relief request is acceptable for PSI based on the examinations performed during fabrication which exceed the VT-3 examination required for PSI.

G. Essential Service Water System Pump Supports, Examination Category D-A (4 Supports)

Supports ID

UEF11 - R006
UEF11 - R007
UEF11 - R008
UEF11 - R0^9

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Code Requirement: Table IWD-2500-1, Examination Category D-A, Item D.1.2, requires component supports and restraints, within the boundary of the above system, for components exceeding 4-inch nominal pipe size, receive a visual examination (VT-3) during each inspection period.

Relief Request: Relief is requested from performing the required preservice VT-3 examination.

Reason for Request: The pump supports are inaccessible because they are submerged within the essential service water pump pit. The supports contain both partial and full penetration welds in each support. The subject weldments were visually inspected during construction; weld material controls, materials traceability and support configurations were also verified by the Callaway constructor during field fabrication of these units. In addition, the piping supports were independently inspected in the course of the Callaway piping systems walkdown performed in response to IE Bulletin 79-14.

Staff Evaluation: The relief request is acceptable for PSI based on the examinations performed during fabrication which exceed the VT-3 examination required for PSI.

IV. CONCLUSIONS

On the basis of the foregoing, pursuant to 10 CFR 50.55a(a)(2), 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 staff technical evaluation has not identified any practical method by which the existing Callaway Nuclear Power Plant can meet all the specific preservice inspection requirements of Section XI of the ASME Code. Requiring compliance with all the exact Section XI required inspections would delay the full-power operation of the plant in order 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 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 the staff review and evaluation, it is concluded that the 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 50.55a(a)(2), relief is allowed 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.

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