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Evaluation of the Watts Bar Nuclear Unit 1 Technical Specifications

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TECHNICAL EVALUATION REPORT
EVALUATION OF THE WATTS BAR NUCLEAR UNIT 1
TECHNICAL SPECIFICATIONS

Docket 50-390

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ABSTRACT

This document was prepared for the Nuclear Regulatory Commission (NRC) to assist them in determining whether the Watts Bar Technical Specifications are in conformance with the assumptions of the Final Safety Analysis Report as amended, the requirements of the Safety Evaluation Report, and Supplemental Safety Evaluation Reports. A comparative audit of the Watts Bar Technical Specifications(TS), Watts Bar Technical Requirements(TRM), Final Safety Analysis Report (FSAR) as amended, Safety Evaluation Report (SER), and Supplemental Safety Evaluation Reports (SSERs) was performed. Identified discrepancies are presented in Section 3 of this report.

SUMMARY

The specific sections of the TS reviewed are listed in section 2. The results of the review is listed in section 3 with a status code assigned each item.

During the performance of this comparative audit review, several differences between TS, TRM, FSAR, SER, and SSERs were noted. A discussion of the differences between sections of the TS, TRM, FSAR and amendments, SER, and SSERs are identified in Section 3.

PREFACE

This report is supplied as part of the "Technical Assistance in Support of Audits on Plant-Specific Conversions to STS" being conducted for the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation by Lockheed Idaho Technologies Company, National Nuclear Operations Analysis Department.

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TECHNICAL EVALUATION REPORT
EVALUATION OF THE WATTS BAR
TECHNICAL SPECIFICATIONS

1. INTRODUCTION

The Watts Bar, Westinghouse Pressurized Water Reactor (PWR) plant, has been selected for a comparison audit review to determine if the Technical Specifications (TS) and Technical Requirements Manual (TRM) are consistent with the Final Safety Analysis Report (FSAR) through Amendment Number 89, and requirements in the Safety Evaluation Report (SER) and the Supplemental Safety Evaluation Reports (SSERs) through SSER-15.

3. CONSISTENCY COMPARISON OF THE WATTS BAR TECHNICAL SPECIFICATIONS AND TECHNICAL REQUIREMENTS MANUAL TO THE FSAR, SER, AND SSERs

3.1 Safety Limits (SLs)

This section covers the review of the safety limits as defined in section 2.0 of the TS. It includes Reactor Core and RCS Pressure limits to ensure that TS agree with or are conservative relative to the FSAR and SER.

Technical Specification	FSAR Section	SER Section	Evaluation
2.1.1 Reactor Core SLs	4.4	4.4.3	DIFFERENT
2.1.2 RCS Pressure SLs	5.2.1.2 Table 5.2-7	Table 5.4-1	CONSISTENT

The reactor core safety limits of the TS (Figure 2.1.1-1) could not be found in the FSAR.

RESOLUTION:

Figure 15.1.1 of the FSAR identifies the Safety Limits. The reactor core safety limits contained in the TS are consistent with the FSAR.

3.2 Reactor Protection System (RPS) Setpoints

This section covers the review of the Reactor Protection System (RPS) setpoints to ensure the TS values agree with or are conservative, relative to the values in the FSAR, the SER and SSERs. As there were no RPS setpoints in the FSAR, the NRC Staff directed that the review be done against WCAP-13721-Westinghouse Setpoint Methodology for Protection System - Watts Bar Units 1 and 2 Eagle 21 Version (Non-Proprietary).

Technical Specifications	FSAR Section	SER Section	Evaluation
3.3.1 RTS Instrumentation	7.2	7.1.3.1, 7.2	DIFFERENT

Observed differences were as follows:

(1) The value for Pressurizer Pressure listed in Table 3-22, page 51, of Ref. 17 disagrees with the value for P′ (nominal RCS operating pressure) used in Table 3.3.1-1 in TS. Ref. 17 uses a value of "$\geq$ 2235 psig - indicated uncertainties." A value of "$\leq$ 2235 psig" is used in TS. Allowing a value less than 2235 psig for P′ would be non-conservative. For example, if a value of 2220 psig was used for P′ then only pressurizer pressure values less that 2220 psig would result in a decrease in the overtemperature ΔT setpoint. FSAR section 7.2.1.1.12 on page 7.2-4 states that P^{o}, (Nominal RCS Operating Pressure) will be "= 2235 psig."

RESOLUTION:

The applicant agreed with the comment and the value in Technical Specifications is changed to be consistent with the FSAR and Ref. 17.

The information is contained in TVA Calculation WBPE680009008 which is available at TVA for audit.

(2) A trip setpoint for Undervoltage RCPs of 90 V was provided in Ref. 17, pages 68 and 69, however, the value provided in TS was $\geq$ 4734 V.

RESOLUTION:

The information is contained in TVA Calculation WBPE680009007 which is available at TVA for audit.

(3) The allowable value for Undervoltage RCPs was not verified because Ref. 17 stated that it was determined by a TVA Calculation.

RESOLUTION:

The information is contained in TVA Calculation WBPE680009008 which is available at TVA for audit.

(4) Time delays associated with SG Water Level Low Low were not included in Ref. 17.

RESOLUTION:

The information is contained in Setpoint and Scaling Document (SSD) 1-L-3-38T which is available at TVA for audit.

(5) Trip setpoints and allowable values associated with Turbine Trips were not included in Ref. 17.

RESOLUTION:

The setpoints are not part of the safety analysis; therefore, the item is resolved based on conversations with the applicant which indicate the information is contained in Westinghouse internal correspondence.

(6) Allowable value and trip setpoint provided in Table 3.3.1-1 of TS for Intermediate Range Neutron Flux, P-6 was not included in the FSAR or Ref. 17.

RESOLUTION:

The information is contained in TVA Calculation 1-NMD-92-131 which is available at TVA for audit.

(8) Allowable values and trip setpoints for Intermediate Range Neutron Flux and Source Range Neutron Flux were not included in Ref. 17.

RESOLUTION:

The information is contained in TVA Calculation 1-NMD-92-131 which is available at TVA for audit.

- SD-1-PS-40-13-S
- SD-1-PS-40-40-S

- 6.f, Motor-Driven Auxiliary Feedwater Pumps Train A and B Suction Transfer on Suction Pressure - Low

RESOLUTION:

The information is contained in TVA Calculation 1-PS-3-139A which is available at TVA for audit.

- 6.g, Turbine-driven AFW Pump Suction Train A and B Transfer on Suction Pressure - Low

RESOLUTION:

The information is contained in TVA Calculation 1-PS-3-121A which is available at TVA for audit.

(3) Allowable values and trip setpoints were not verified for the following items listed on page 3.3-39 of TS because Ref. 17 provided values in percent of span and TS uses inches from a specific reference:

- 7.b, Refueling Water Storage Tank (RWST) Level - Low Low

RESOLUTION:

The values are contained in TVA Calculation 1-L-63-50 which is available at TVA for audit.

- 7.b, Containment Sump Level - High

RESOLUTION:

The values are contained in TVA Calculation 1-L-63-180 which is available at TVA for audit.

(4) Trip setpoints and allowable values for 8.b Pressurizer Pressure on page 3.3-39 were not verified because values were not provided in Ref. 17.

RESOLUTION:

The values are contained in TVA Setpoint and Scaling Document SSD-1-T-68-323 which is available at TVA for audit. The values are also contained in the Precautions, Limitations and Setpoints document supplied by Westinghouse to TVA.

3.4 Containment Isolation Valves (CIVs)

This section covers the review of all the CIVs to ensure that the TS agree with or are conservative relative to the FSAR and SER.

3.7 Combustible Gas Control System LCOs

This section covers the review of the LCOs identified for Containment Gas Control in the FSAR and SER to ensure they are included in TS.

Technical Specifications	FSAR Section	SER Section	Evaluation
3.3.2 Post Accident Monitoring (Containment Hydrogen Concentration)	6.2.5	6.2.5	CONSISTENT
3.6.7 Hydrogen Recombiners	6.2.5	6.2.5	DIFFERENT
3.6.8 Hydrogen Mitigation System	6.2.5A	6.2.5	CONSISTENT

Section 6.2.5.2, page 6.2.5-3, of the FSAR specifies that the hydrogen recombiners heat air to 1150 to 1400 ° F, where recombination of hydrogen and oxygen occurs. The Bases discussion for SR 3.6.7.1, page B 3.6-47, only requires that a temperature of 700 ° F be achieved by the surveillance. While the SR does require power to be increased to maximum, the SR does not verify that temperature is increased to a value that would result in recombination.

RESOLUTION:

Applicant committed to revising the FSAR to discuss how the Hydrogen Recombiners are tested.

3.8 Ice Condenser LCOs

This section covers the review of the LCOs identified for Ice Condenser in the FSAR and SER to ensure they are included in TS.

Technical Specifications	FSAR Section	SER Section	Evaluation
3.6.11 Ice Bed	6.7 6.2.1	6.2.11	DIFFERENT
3.6.12 Ice Condenser Doors	6.7	6.2.1	CONSISTENT

The following differences were observed:

(1) FSAR section 6.7.4.1, page 6.7-14, indicates that the baskets contain a minimum of 2.125×10^6 pounds of ice. SR 3.6.11.2, page 3.6-29, requires that the total weight of stored ice be $\geq$ 2,360,875 lb. This is CONSERVATIVE.

(2) TS SR 3.6.11.4, page 3.6-29, specifies that the accumulation of ice or frost on structural members comprising flow channels through the ice condenser is $\leq$ .38 inches thick. The Bases discussion for TS SR 3.6.11.4, page B 3.6-71, identifies that the value is based on the analysis of containment response to a DBA with partial flow blockage. The comparative audit was unable to locate the value used in TS in the FSAR.

Tennessee River flood that could inundate the Watts Bar plant site. Specific details of the plan--such as target river levels, initiation of Stage I and Stage II shutdown, communications--are incorporated in a Technical Specification and limiting condition for operation.

TRM Section: 3.7.2 pages 3.7-3 through 3.7-9.

This requirement was located in the TRM instead of the TS as part of the adoption of the Improved Technical Specifications (ITS).

The SER in section 2.4.3 indicates that shutdown procedures will begin at a specific river level but the TRM indicates that shutdown procedures will begin based at specific flood stage. The documents provided did not provide sufficient data to determine if the river levels specified in SER section 2.4.3 are consistent with the flood stages specified in TRM.

This item is DIFFERENT.

RESOLUTION:

FSAR section 2.4.14.8.2 on page 2.4-57 correlates the relationship between projected river levels and Flood Stages. The information in the TRM is consistent with the levels stated in the SER.

2. SER Section 3.9.6, Inservice Testing of Pumps and Valves, on page 3-32 states:

There are several safety systems connected to the reactor coolant pressure boundary that have design pressure below the rated reactor coolant system (RCS) pressure. There are also some systems which are rated at full reactor pressure on the discharge side of pumps but have pump suction below RCS pressure. In order to protect these systems from the RCS pressure, two or more isolation valves are placed in series to form the interface between the high-pressure RCS and the low-pressure systems. The leaktight integrity of these valves must be ensured by periodic leak testing to prevent exceeding the design pressure of the low-pressure systems and thus causing an intersystem LOCA. Periodic leak testing of pressure isolation valves shall be performed after all disturbances to the valve are complete. The pressure isolation valves to be tested are listed in the Technical Specifications.

TS Section: None

This requirement was located in the FSAR table 3.9-17 instead of the TS as part of the adoption of the Improved Technical Specifications (ITS). Refer to section 3.6 of this report for concerns with the listing of pressure isolation valves to be tested.

This item is CONSISTENT.

3. SER section 3.9.6, Inservice Testing of Pumps and Valves, on page 3-32 states:

The staff requires that the Watts Bar Technical Specifications contain limiting conditions for operation that will require plant shutdown or system isolation when the leakage limits are not met. The Technical Specifications will include surveillance requirements which state the acceptable frequency of leak rate testing and the acceptable value for leakage rate. The Technical Specifications as discussed in the section will be based on the latest revision of NUREG-0452, "Standard Technical Specifications for Westinghouse Pressurized Water Reactors." However, the Technical Specification for the Watts Bar facility have not yet

By letter dated August 24, 1992 (submittal regarding use of V5H fuel design), the applicant addressed fuel rod bowing issues. The maximum rod bow penalties (< 1.5% DNBR) accounted for in the design safety analysis are based on an assembly average burnup of 24,000 MWD/MTU. A 10.7-percent DNBR margin is maintained for the V5H fuel by comparing the DNBR limit of 1.31 to the WRB-1 correlation limit of 1.17. The applicant has incorporated the DNBR margin and residual rod bowing penalty into the "Bases for Safety Limits" section (Section 2.0) of the draft TSs. In the August 24, 1992, letter, the applicant stated that Table 4.1 ("Rod Bow Penalties") of the SER no longer applies; the staff concurs with the applicant in light of discussion in this paragraph.

The staff reviewed the applicant's response and finds it acceptable since an approved method (WCAP-8762-P-A, "New Westinghouse Correlation WRB-1 for Predicting Critical Heat Flux in Rod Bundles with Mixing Vane Grids," July 1984) was used, and the DNBR margin and all rod bow penalties have been incorporated into an appropriate section of the TSs. This effort was tracked by TAC M76742.

TS Section: B 2.1.1 page B 2.0-4

The Bases discussion for Safety Limit 2.1.1, page B 2.0-4, discusses application of the rod bow penalty to the DNBR safety limit.

This item is CONSISTENT.

5. SER section 4.2.3, Mechanical Performance, on page 4-7 states:

Although revised Westinghouse cladding models have been reviewed and accepted the staff's review of the new ECCS evaluation model has not been completed. Therefore, the staff requested that supplemental calculations be performed for the ECCS analyses in the Watts Bar FSAR with the materials models of NUREG-0630 unless a new ECCS model is available. In a letter dated September 22, 1981, the applicant submitted the results of the supplemental ECCS calculations which show that Watts Bar will continue to meet the acceptance criteria of 10 CFR 50.46 if the F_Q peaking factor is reduced to 2.31. Consequently, the Technical Specifications must reflect the new Watts Bar F_Q value of 2.31. The staff, therefore, concludes that the issue of cladding swelling and flow blockage has been resolved for Watts Bar.

TS Section: 3.2.1 page 3.2-1

Heat Flux Hot Channel Factor ($F_Q(Z)$) is specified in the Core Operating Limits Report (COLR) which was not provided for the comparative audit so the actual value was not verified.

The actual value was not verified in COLR but this item is considered CONSISTENT.

6. SER Section 4.4.4.2, Crud Deposition, on page 4-18 states:

Crud deposition in the core and an associated change in core pressure drop and flow have been observed in some PWRs not of Westinghouse design. In response to a staff question, the applicant stated that (1) operating experience on Westinghouse reactors indicate that a flow resistance allowance for crud deposition is not required; (2) the effects of crud enters into the calculations by use of a surface roughness factor three times greater than those obtained from operating Westinghouse PWRs; and (3) reduced flow would be observed by four different means: (1) flow meters, (2) a reduction in reactor power if in an automatic control mode or an increase of ΔT across the core if in a manual mode, (3) core exit thermocouple readings, and (4) incore flux maps.

has shown that the LPMS meets Positions I through 4 in Section C of RG 1.133 regarding the LPMS characteristics, alert level establishment, data acquisition, and safety analysis report content. With regard to Position 5, the limiting conditions for operation and surveillance requirement with respect to the LPMS have been incorporated into Technical Specifications 3.3.3.11 and 4.3.3.11. Position 6, "Notification of a Loose Part," has not been addressed by the applicant. However, 10 CFR 50.73 requires that the applicant shall submit a licensee event report (LER) within 30 days after the discovery of "any event or condition that resulted in the condition of the nuclear power plant, including its principal safety barriers, being seriously degraded, or that resulted in the nuclear power plant being: (a) in an unanalyzed condition that significantly compromised plant safety and (b) in a condition that was outside the design basis of the plant." Because the presence of loose parts in the primary system belongs to this category, an LER would be required according to 10 CFR 50.73. Because 10 CFR 50.73 became effective January 1, 1984, and replaces all existing requirements for applicants to report reportable occurrences as defined in individual plant Technical Specifications, Position 6 is covered by this regulation.

TRM section: 3.3.6 pages 3.3-20 and 3.3-21.

This requirement was located in the TRM instead of the TS as part of the adoption of the Improved Technical Specifications (ITS).

This item is CONSISTENT.

8. SER Section 4.4.7, N-1 Loop Operation, on page 4-20 states:

N-1 loop operation is when one of the reactor's coolant loops is out of service. Thus, only three coolant loops are available to supply coolant to the reactor core.

In response to a staff question, the applicant stated that he did not wish to exercise the option to operate in the N-1 mode. The staff will require that the Technical Specifications include appropriate provisions to ensure that this type of operation is prohibited.

TS Section: 3.4.4 page 3.4-7

Four loop operation is required in Modes 1 or 2 per TS section 3.4.4.

This item is CONSISTENT.

9. SER Section 5.2.2, Overpressure Protection, on page 5-3 states.

With regard to operation when the RCS is at low temperature (during startup and shutdown), the staff is concerned about potential reactor vessel damage as a result of an overpressure transient. The applicant has provided a description of the low temperature overpressure protection design, describing a system that, once manually-enabled, will automatically monitor RCS pressure and temperature and will provide backup control of the pressurizer PORVs. Proposed administrative procedures to reduce the potential for low temperature overpressure events also were addressed. Additionally, the applicant has discussed design low temperature overpressure transients and further details on the administrative procedures associated both with operation of the system and with reducing the potential for overpressure events. The applicant has committed to provide plant-specific analyses, which will dictate governing Technical Specifications.

TS Section: 3.4.12 pages 3.4-25 through 3.4-29

This difference was resolved by SSER 12.

This item is CONSISTENT.

11. SER Section 6.2.1.1, Containment Structure, on page 6-6 states:

Based on the above discussion of current ice condenser operating experience, the staff will require the applicant to institute a periodic ice basket weighing program for each unit at Watts Bar, similar to the program being conducted at the Sequoyah plant. The staff has recommended that the applicant continue to evaluate the equipment and techniques available for ice loading to achieve an initial ice inventory that is uniformly distributed. The staff will pursue the development of a suitable periodic ice weighing program with the applicant during the development of Technical Specifications for the operation of the plant and will include appropriate operating limits to ensure an acceptable margin of safety.

TS Section: 3.6.11 on page 3.6-29

Periodic ice weighing is required by SR 3.6.11.2 and SR 3.6.11.3.

This item is CONSISTENT.

12. SER Section 6.2.1.1, Containment Structure, on page 6-6 states:

Based on the above discussion of current ice condenser operating experience, the staff will require the applicant to institute a periodic ice basket weighing program for each unit at Watts Bar, similar to the program being conducted at the Sequoyah plant. The staff has recommended that the applicant continue to evaluate the equipment and techniques available for ice loading to achieve an initial ice inventory that is uniformly distributed. The staff will pursue the development of a suitable periodic ice weighing program with the applicant during the development of Technical Specifications for the operation of the plant and will include appropriate operating limits to ensure an acceptable margin of safety.

SSER 5 Section 6.2.1.1, Containment Structure, on page 6-1 states:

The revised analysis is based on 2.125×10^6 lb of ice initially being in the ice condenser, instead of 2.45×10^6 lb, which was assumed in the original analysis. Therefore, the staff concludes that the lower ice weight may be used as the basis for establishing the technical specification limit for the minimum allowable weight of ice in the ice condenser. This review was tracked by licensing action TAC 63620.

TS Section: B 3.6.11 page B 3.6-67

The Bases discussion for 3.6.11 refers to FSAR section 6.2, which on page 6.2.1-5, indicates that the input assumption for ice in the ice condenser is 2.125×10^6 lbs. of ice. SR 3.6.11.1 provides a limit of $\geq$ 2,360,875 lb which is conservative.

This item is CONSERVATIVE.

13. SER Section 6.2.6, Containment Leakage Testing, on page 6-22 states:

The applicant has designed the Watts Bar plant containments so that there is no potential path by which containment leakage could bypass both the emergency gas treatment system and the auxiliary building gas

The comparative audit performed a comparison of the values in the TS and the Setpoint Methodology in sections 3.2 and 3.3 of the report. Identification of inconsistencies that exist in the final draft are identified in sections 3.2 and 3.3 of this report.

This item is CONSISTENT, with the exception of items identified in section 3.2 and 3.3 of this report.

15. SER Section 7.2.3, Testing of Reactor Trip Breaker Shunt Coils, on page 7-5.

A related concern involves the testing of the manual reactor trip switch operation. Each of the switches controls power to both the undervoltage coil and the shunt trip coil of each breaker. De-energizing the undervoltage coil or energizing the shunt coil trips the breaker. As a result, tripping a breaker when testing a manual switch does not confirm that both breaker trip coils operate as intended. The staff expressed its concern to the applicant and advised him that the staff will require in the Technical Specifications that the shunt coils be tested at least once each refueling outage.

TS Sections: SR 3.3.1.4 page 3.3-11
SR 3.3.1.13 page 3.3-14.

These surveillances verify operation of the shunt coils at a frequency of 31 days.

This item is CONSERVATIVE.

16. SER Section 7.6.5, Overpressure Protection During Low Temperature Operation, on page 7-17 states:

The overpressure protection during low temperature operation is provided by automatic actuation of the pressurizer PORVs. In its review of the automatic control logic for these valves, the staff found that a failure resulting in a high output signal from either of the two auctioneers would prevent both of the valves from opening when needed. Because no indication of the auctioneered safety output signals is provided to the operator, such a failure could remain undetected. Furthermore, even if such a failure would be detected, the system would remain inoperative because no capability to manually arm the system to replace a failed permissive signal from the auctioneer is provided. By letter dated February 12, 1982, the applicant has committed to install switches on the main control board for the operator to manually arm this system. The manual arming will be included in the operating procedure when the reactor coolant temperature is equal to or below the setpoint and before beginning the filling operation. The system arming will be reset when the system is brought back above the system temperature setpoint for arming. The staff will define the arming setpoint in the Technical Specifications, and the instruments for over-pressure protection will be under periodic surveillance test. The staff finds this design acceptable subject to its review of the updated drawings and FSAR descriptions to be submitted by the applicant. The staff will report on its final conclusions in a supplement to the SER.

SSER 4 Section 7.6.5, Overpressure Protection During Low-Temperature Operation, on pages 7-1 and 7-2 states:

By Amendment 52 of the FSAR, the applicant provided updated information on the reactor coolant system (RCS) pressure control system used during low-temperature operation. Two power-operated relief valves (PORVS) are used to provide overpressure protection of the RCS during low-temperature operation. The PORVs are automatically opened when RCS pressure exceeds a programmed setpoint based on RCS temperature. During normal operation, this system is manually blocked to preclude a single failure

18. SER Section 8.3.2.5, Nonsafety Loads Powered from the DC Distribution System and Vital Inverters, on page 8-14 states:

The applicant has stated in Section 8.3.2.1.1 of the FSAR that the batteries have the capacity to supply all connected loads (Class 1E and non-Class 1E) for a minimum of 2 hours and that the batteries will be tested periodically in accordance with the Technical Specifications to ensure this capacity. This meets review guidelines and is acceptable.

TS Section: SR 3.8.4.13 and SR 3.8.4.14 on pages 3.8-28 and 3.8.29.

The listed surveillance requirements test the battery capacity.

This item is CONSISTENT.

19. SER Section 8.3.3.1.1, Submerged Electrical Equipment as a Result of a Loss-of-Coolant Accident, on page 8-15 states:

With the exception of circuits to be analyzed, the above described relocations and design provisions meet the staff position and are acceptable. Implementation of automatic de-energization design provisions will be verified as part of the site visit/drawing review. Surveillance requirements for components with power lockout provisions will be pursued as part of the Technical Specifications review.

In regard to power systems functional aspect, the resolution regarding submergence of equipment is acceptable pending incorporation of surveillance requirements for the protective devices in the Technical Specifications.

SSER 13 Section 8.3.3.1.1, Submerged Electrical Equipment as a Result of a Loss-of-Coolant Accident, on pages 8-19 and 8-20 states:

In the SER and SSER 3, the staff stated that the design for the automatic deenergizing of loads as a result of a loss-of-coolant accident (LOCA) would be verified as part of the staff's site visit/drawing review. This activity was tracked by Confirmatory Issue 42. By letter dated January 16, 1985, the applicant submitted electrical schematic drawings for each valve identified to be automatically deenergized by a LOCA signal. On the basis of a review of drawing 45W760-30-8, Revision 10, the staff concluded that the design for power removal was acceptable pending resolution of surveillance requirements that were to be reviewed with and included in the Technical Specifications.

In the SER, the staff similarly stated that the design using power lockout or protective devices was acceptable pending resolution of surveillance requirements that were to be reviewed with and included in the Technical Specifications.

Subsequently, as part of a technical specifications improvement program, the staff concluded that such surveillance requirements were not appropriate for inclusion in technical specifications, but should be reviewed with and included in the FSAR as licensing commitments. In order to initiate its review of the required surveillance, the staff asked the applicant for additional information in its RAI dated June 20, 1991.

During the site review of August 7 and 8, 1991, and in subsequently in a letter dated September 13, 1991, the applicant stated that it would revise the FSAR to state that the components listed in Table 8.3-28 are

SSER 13 Section 8.3.3.2.1, Sharing of DC Distribution Systems and Power Supplies Between Units 1 and 2, on page 8.22 states:

In the SER the staff stated that the dc system for Unit 1 supplies power to vital buses I and II for Unit 2, and the dc system for Unit 2 supplies power to vital buses III and IV for Unit 1. This was wrong; the Unit 1 dc system supplies Unit 1 buses I and 11 and the Unit 2 dc system supplies Unit 2 buses III and IV. Because the dc distribution system design provides for normal and alternate supplies to system boards and some components, some sharing of the dc system can occur in certain modes of operation. See Section 8.3.2.5.1 herein for a discussion of the acceptability of this design feature.

Also in the SER, the staff concluded (based on results of an analysis submitted in a letter dated January 7, 1982, imposition of appropriate Technical Specifications for shared dc systems, and other information documented in the FSAR) that there was reasonable assurance that the sharing would not significantly impair the ability of the dc system to perform its safety function, meets the requirements of GDC 5, and was acceptable. However, after further review of the results of the analysis submitted in the letter dated January 7, 1982, and information documented in the FSAR, it was not clear that the sharing of raceways would not significantly impair the ability of the dc system from performing its safety function for any given single failure in accordance with the requirements of GDCs 5 and 17. The staff was concerned that a single failure of the shared train A (or train B) raceway may cause failure of two dc vital battery systems. The staff told the applicant about these two concerns in the RAI dated June 20, 1991.

TS Sections: 3.8.4 and 3.8.5

The SER requirement was based on two unit operation. Based on operation of only Unit 1 this item is resolved.

21. SER Section 8.3.3.2.2, Sharing of AC Distribution Systems and Standby Power Supplies Between Units 1 and 2, on page 8-17 states:

In Section 8.3.1.1 of the FSAR, the applicant states that the standby power system serving each unit is divided into two redundant power trains with loads connected so that failure of an entire train will not prevent the redundant train from performing the required safety function. The applicant was asked to provide a description of how the design satisfies position C2 of Regulatory Guide 1.81. By letter dated October 9, 1981, the applicant provided the description of compliance. This description appears to contradict Section 8.3.1.1 of the FSAR in that Section 8.3.1.1 implies two diesels (one per unit) can be lost rather than only one diesel out of four, as stated in the October 9, 1981 letter. Similarly, Section 8.3.1.1 contradicts Sections 8.1.4 and 3.1.2.1 of the FSAR. Sharing of onsite power systems (ac and dc) had not been adequately described or analyzed in Section 8.3 of the FSAR as required. For example the essential raw cooling water (ERCW) that is needed to mitigate an accident in Unit 1 is dependent on pumps powered from Unit 2 to meet the single-failure criterion. This sharing of ac power systems had not been described or analyzed in Section 8.3 of the FSAR. By letter dated January 7, 1982, the applicant provided the subject description and analysis. Based on this letter and imposition of appropriate Technical Specifications for shared ac power systems, the staff concludes that the design meets the guidelines of Regulatory Guide 1.81 and is acceptable pending revision of the FSAR that reflects requirements of the shared safety systems. Surveillance requirements for sharing of ac systems will be pursued as part of the Technical Specifications review.

TS Sections: 3.8.1 and 3.8.2

material regarding testing. In the proposed Technical Requirements Manual (submitted as part of the Technical Specifications review effort, TAC M76742), the applicant has included surveillance requirements (functional testing) for circuit breakers which are used as isolation devices. On the basis of the review and acceptance of that document, this issue is resolved.

TRM Section: 3.8.1

Surveillances requirements are incorporated in the TRM that comply with the requirements of the SER.

This item is CONSISTENT.

23. SER Section 8.3.3.6, Compliance with GDC 50, on page 8-24 states:

In regard to testability of fuses, the applicant has indicated that fuses will be periodically checked for proper size, deterioration, and the connection's tightness and cleanliness. In addition, the staff will require, and include as part of its Technical Specification, periodic measurement of the fuses an terminal connection resistance.

SSER 13 Section 8.3.3.6, Compliance with GDC 50, on pages 8-28 states:

In FSAR Section 8.1.5.3, the applicant stated, as an exception to the testability requirements of RG 1.53, that, in lieu of testing of fuses by resistance measurement, a fuse inspection and maintenance program would be established. Because justification was not presented in the FSAR for this exception, the staff stated in the SER that periodic resistance measurement of fuses and their terminal connections would be required as part of the Technical Specifications.

By letter dated September 15, 1982, and FSAR Amendment 55, the applicant submitted information as to why periodic resistance measurement is not practical and provided justification for the adequacy of the proposed inspection and maintenance program in lieu of resistance measurement testing. However, the applicant, by FSAR Amendment 63, removed this justification. Therefore, on the basis of information presented in the FSAR, the staff was unable to find the applicant's proposed inspection and maintenance program acceptable.

In regard to the statement in the SER that periodic resistance measurement of fuses and their terminal connections would be required as part of the Technical Specifications, the staff has subsequently concluded as part of the standard technical specifications improvement program that this type of testing requirement should not appear in the plant's Technical Specifications but should be included instead in the FSAR as a design-basis requirement for the plant. Periodic resistance measurement of fuses and their terminal connections was, therefore, not required as part of the Technical Specifications as indicated in the SER, but was addressed as part of the staff's review of the FSAR. The staff told the applicant of its concerns pertaining to the adequacy of the proposed inspection and maintenance program in the RAI dated June 20, 1991.

TS Section: None

The SSER withdrew the concern with fuse measurement identified in the SER.

This item is CONSISTENT.

(2) exercising and observing of the main steam stop and control, reheat stop and intercept valves at least once a week. The applicant has not committed provide the above inservice inspection program. The staff requires the total program and will include this requirement in the Technical Specifications.

SSER 5 Section 10.2, Turbine Generator, on page 10-1 and 10-2 states:

In letters dated March 7 and June 19, 1984, and March 25, 1985, and at meetings in NRC offices on January 22, February 6, and February 28, 1985, the applicant requested the deletion of Standard Technical Specification (STS) 3/4.3.4, "Turbine Overspeed Protection," for Watts Bar Units 1 and 2. As a result of these meetings and discussions with the applicant, the applicant, in letters dated March 25 and April 9, 1985, modified its request so that the surveillance requirements for the turbine integrity program with turbine overspeed protection (TIPTOP) would become part of the administrative controls (Section 6.8.5) of the Technical Specifications and the turbine valve testing frequency would change from once a week to once a month.

The staff has evaluated the information submitted by the applicant in the March 25 and April 9, 1985, letters and Westinghouse information presented at the March 23, 1983, meeting. Considering the information presented by the applicant and Westinghouse and the staff's original basis for the STS, the staff concludes that the interval between periodic turbine valve testing can be increased for Watts Bar from weekly to monthly without significantly affecting the capability of the turbine valves to function on demand. In summary, the basis for considering Technical Specifications relief at Watts Bar was:

(1) Provision of a satisfactory statistical basis to determine frequency of turbine valve testing. Up to now, test frequency of these valves has been largely based on experience with turbine generators installed in fossil-fuel plants. The Westinghouse turbine-missile study (Westinghouse, March 1974) provides a basis for Westinghouse to establish turbine valve test frequency on a monthly basis for all nuclear units having a steam chest and valve arrangement similar to that at Watts Bar.

(2) The applicant's maintenance, inspection, and turbine valve test program described in the FSAR and the proposed modification to the Technical Specifications appear to be satisfactory. This program performed on a periodic basis coupled with monthly testing of all turbine valves is satisfactory to the staff.

(3) The data and rationale presented by Westinghouse at the March 23, 1983, meeting and earlier meetings with the staff and the staff's understanding of the data presented to date, including WCAP-11525.

(4) Testing of turbine control valves on base-loaded machines necessitates reducing generator output for several hours. The valve testing sequence during turbine operation requires placing the turbine on manual control and repositioning all turbine control valves in the steam chest to permit individual full-valve stroking. All valves are aligned to equal position. Repositioning the control valves (on a base-loaded machine) results in reduced steam flow to the turbine with a consequent reduction in generator output of about 5 percent. All turbine control valves are tested in a relatively short time (about 35 to 40 minutes). The bulk of the time (approximately 21/2 to 3 hours) is consumed in slowly lowering reactor output to permit control valve testing. On completion of valve tests, a similar time period is consumed in slowly increasing reactor power to permit full-load operation of the turbine generator. Reactor output must be changed slowly to minimize xenon spiking. Although this economic impact is not a safety consideration, the staff factored it into its action.

Report requirements on the fire detection, carbon dioxide, sprinkler, and fire-barrier systems. Furthermore, procedural controls require that a fire watch be established anytime a fire door is breached (which would occur when routing a hose to the refilling location in the DG buildings). Therefore, a separate and distinct fire watch would not be required during the refilling operation. The fire door watch provides the necessary additional protection.

On this basis, the staff concludes that the fire watch as described in Section 9.5.4.2 of the SER is not required and that the alterative refilling method is acceptable. This effort was tracked by TAC M76742.

TS Section: None

The requirement identified in the SER for a fire watch was removed by the SSER.

The Technical Requirements Manual or Technical Specifications submitted by the applicant do not contain requirements concerning fire protection.

This item is CONSISTENT.

28. SER Section 10.3.4, Secondary Water Chemistry, on page 10-6 states:

Because of the complexity of the corrosion phenomena involved and the state of the art as it exists today, the staff is of the opinion that, in lieu of specifying limiting conditions in the Technical Specifications, a more effective approach would be to institute a license condition that required the implementation of a secondary water chemistry monitoring and control program containing appropriate procedures and administrative controls.

SSER 5 Section 10.3.4, Secondary Water Chemistry, on page 10-2 states:

In the SER, the staff found the applicant's secondary water chemistry monitoring and control program met requirements. The staff further stated that the license will be conditioned to require that the program be implemented. The Technical Specifications are being developed for Watts Bar Unit 1 and this program is being included in the administrative section. This will ensure implementation of the program and eliminates the need for a license condition. Thus, proposed License Condition 23 is considered resolved.

TS Section: 5.7.2.13 pages 5.0-21 and 5.0-21.

Section 5.7.2.13 of the TS contains the requirements for a Secondary Water Chemistry Program which meets the requirements of the SER.

This item is CONSISTENT.

29. SER Section 10.4.4, Turbine Bypass System, on page 10-9 states:

The applicant will include preoperational and startup tests of the turbine bypass system in accordance with recommendations of Regulatory Guide 1.68. The adequacy of the test program is evaluated in Section 14.1 of this report. The turbine bypass system can be tested while the unit is on line. The applicant's proposed inservice inspection program will provide for stroking the valves every fueling outage and a visual external inspection every refueling outage. The staff finds this unacceptable. The staff requires, as

flattening. The specific level of prepressurization will depend on the planned fuel burnup and will be determined before the Technical Specifications are established.

SSER 13 Section 4.2.1, Description, on page 4-1 states:

The staff reviewed the applicants's response, and agrees that the requested information, already documented in SSER 2, is not key design information that needs to be included in the Technical Specifications. This effort was tracked by TAC M76742.

TS Section: None

The SSER resolved the concern identified in the SER.

This item is CONSISTENT.

32. SER Section 4.2.4, Surveillance, on page 4-8 states:

At the Watts Bar facility, performance of the fuel is indirectly monitored by measurement of the activity of the primary coolant for compliance with Technical Specification limits. Westinghouse has proposed a fuel surveillance program for several plants that will use the 17 x 17 fuel assemblies. A summary of this program is given in the fuel rod bowing report, WCAP-8692. This program includes lead assemblies in Surry Units 1 and 2 and the initial core loadings for Trojan, Beaver Valley Unit 1, Farley Unit 1, and Salem Unit 1.

TS Section: 3.4.16

This item is CONSISTENT.

33. SER section 4.3.2.4, Control Rod Patterns and Reactivity Worths, on page 4-12 states:

The full-length control rods are divided into two categories--shutdown rods and regulating rods. The shutdown rods are always completely out of the core when the reactor is at operating conditions. Core power changes are made with regulating rods which are nearly out of the core when it is operating at full power. Regulating rod insertion will be controlled by power-dependent insertion limits required in the Technical Specifications to ensure that:

(1) there is sufficient negative reactivity available to permit rapid shutdown of the reactor with adequate margin

(2) the worth of a control rod that might be ejected is not greater than that which has been shown to have acceptable consequences in the safety analyses

TS Section: B 3.1.7 page B 3.1-41 and B 3.1-42.

The Bases discusses that the power dependent insertion limits will assure that the two requirements listed are complied with.

This item is CONSISTENT.

The first part is discussed in this section under NPSH and sump tests. The second part, sump debris, must be resolved by addressing four objectives:

(1) containment, insulation, and sump design
(2) housekeeping
(3) alarms to alert the operator to ECCS degradation
(4) actions to be taken in the event of ECCS degradation

The second objective is to ensure by appropriate housekeeping procedures that the containment "as-licensed" cleanliness is maintained. The applicant has cited Technical Specifications which address this concern. The staff finds this acceptable.

TS Section: None

The requirement to maintain containment cleanliness is not contained in the final draft TS or TRM. This requirement is not contained in the improved Technical Specifications of NUREG-1431.

This item is CONSISTENT.

38. SER Section 6.5.4, Ice Condenser as a Fission Product Removal System, on page 6-36 states:

The ice condenser is designed to remove iodine from the postaccident atmosphere passing through the ice beds. Sodium tetraborate is added to the ice to enhance the iodine adsorption characteristics of the ice. Technical Specifications require a minimum ice pH of 8.5 whenever the reactor is critical.

TS Section: SR 3.6.11.5 page 3.6-30.

SR 3.6.11.5 requires that ice pH be maintained $\geq$ 9.0 and $\leq$ 9.5.

This item is CONSERVATIVE.

39. SER Section 7.1.3.2, Response Time Testing, on page 7-2 states:

To ensure that the response times of each protective function of the reactor trip system (RTS) and the engineered safety features actuation system (ESFAS) are within the limits assumed in the accident analysis, Technical Specifications require testing the time response at specified intervals. The periodic test frequency specified in the Technical Specifications, as stated by the applicant, is developed primarily on the basis of past operating experience. If the response time of any protection channel were found to be marginal or to require frequent adjustments of instrumentation because of changes on plant conditions, the test frequency would be increased.

TS Section: SR 3.3.1.15, page 3.3.1.15, and SR 3.3.2.10, page 3.3-33.
TRM sections: 3.3.1 and 3.3.2

The TS provide reference to the listed TRM sections which contain actual response times.

This item is CONSISTENT.

sound in the reactor control room, alerting the operator to the condition. Flow will then be diverted from the gas decay tank being filled to a standby tank and a nitrogen diluent introduced into the system to reduce the potential for a hydrogen explosion. The hydrogen and oxygen monitoring system does not meet the staff's current acceptance criteria because redundant monitors are not provided and because the system is not designed to automatically initiate action to mitigate the potential for explosion in the event of a high oxygen content; therefore, the staff will provide a Technical Specification which will require sampling and analysis every 4 hours during gas monitor outages and will require that the reactor be shut down if the gas monitor outage exceeds 7 days. With the inclusion of the Technical Specification described above, the staff finds the system acceptable.

SSER 8 Section 11.3, Gaseous Waste Management, on page 11-1 states:

By letter dated January 25, 1983, the applicant stated that there exists a significant safety concern with having a technical specification that requires that the reactor be shut down within a specified time when the hydrogen or oxygen monitors are out of service. This would lead to degassing at a time when both monitors are inoperable. The staff has evaluated this requirement and concludes that an equivalent level of protection (rather than reactor shutdown) against the uncontrolled release of radioactive materials would be provided for Watts Bar by a technical specification. This would require that grab samples be collected at least once during a batch transfer to a waste gas decay tank and at least once every 4 hours and analyzed within the following 4 hours when a hydrogen or oxygen monitor is inoperable. A special report will have to be submitted to the staff when either monitor is inoperable for more than 7 days or both monitors are inoperable. These requirements will be considered for incorporation into the Watts Bar Technical Specifications, currently being developed.

TS Section: 5.7.2.15 pages 5.0-25 and 5.0-26.

The TS provides that a program be established for Explosive Gas and Storage Tank Radioactivity Monitoring. The program is required to include establishment of limits on hydrogen and oxygen and a surveillance program is required.

This item is CONSISTENT.

43. SER Section 11.5, Process and Effluent Radiological Monitoring and Sampling Systems, on page 11-7 states:

The Technical Specifications for the process effluent radiological monitoring systems, instrumentation, controls, and the sampling and analysis programs for Watts Bar will be similar to those at Sequoyah.

TS Section: 3.3.2, 3.3.6, 3.3.7 and 3.3.8.

A review of the TS submitted by Watts Bar determined that some items listed in Table 11.2 of the SER were not included in TS. The improved Technical Specifications of NUREG-1431 only include monitoring systems that actuate a system included in the TS.

This item is CONSISTENT.

44. SER Section 13.4, Review and Audit, on page 13-28 and 29 states:

Additional surveillance will be required to ensure rod alignment if one or more rod position channels are out of service.

TS Section: 3.1.8 page 3.1-17.

Action A of LCO 3.1.8 provides additional requirements for rod position channels being out of service.

This item is CONSISTENT.

47. SER Section 15.4.2, Main Steamline Break Outside Containment, on page 15-17 states:

For Case 2, the staff assumed that an iodine spike occurred as a result of the power and pressure transient caused by the accident. Prior to the accident, the Watts Bar plant was assumed to be operating at the equilibrium primary coolant Technical Specification limit of 1 ~Ci/gram dose equivalent iodine-131 (DEI-131). The iodine spike generated during the accident is assumed to increase the release rate of iodine from the fuel by a factor of 500. This increase in the release rate results in an increasing iodine concentration in the primary coolant during the course of the accident. The radiological consequences for this case have been calculated using assumptions found in Table 15.3, and the consequence values are reported in Table 15.1.

For Case 3, the staff assumed that previous reactor operation has resulted in primary coolant concentration equal to the maximum transient full power Technical Specification limit (60 uCi/gram DEI-131). As in Case 2, the radiological consequences were calculated using assumptions found in Table 15.3, and the consequence values are reported in Table 15.1.

Based on its findings, the staff concludes that the distances to the exclusion area and low population zone outer boundaries for the Watts Bar site are sufficient to provide reasonable assurance that the calculated radiological consequences of a postulated main steamline failure outside the Watts Bar containment do not exceed (1) the exposure guidelines as set forth in 10 CFR 100.11 for the case in which failure occurs with a primary coolant iodine concentration corresponding to a pre-accident iodine spike; and (2) 10 percent of these exposure guidelines for the case that the failure occurs with a primary coolant activity corresponding to the maximum equilibrium concentration for continued full power operation as stated in the Watts Bar plant-specific Technical Specifications. The staff has reviewed the Watts Bar plant-specific Technical Specifications to ensure that the assumptions used in the analyses identified above are not exceeded. The staff also concludes that the proposed design and operation of the Watts Bar nuclear station is effective in controlling the release of fission products following a postulated main steamline break accident.

These conclusions are based on (1) the staff review of the applicant's analysis of the radiological consequences, (2) the independent dose calculation by the staff using appropriate regulatory assumptions and conservative atmospheric diffusion factors as discussed in Chapter 2 of this report, and (3) the Watts Bar plant-specific Technical Specifications for the iodine concentration in the reactor coolant consisting of a maximum allowable limit and a limit for the equilibrium concentration for continued plant operation and the Watts Bar Technical Specification limit on primary to secondary leakage in the steam generators.

TS Section: 3.4.13 and 3.4.16

The TS contained limits for primary to secondary leakage in the steam generators and limit on equilibrium primary coolant dose equivalent I-131.

the applicant's assumptions and analyses of the radiological consequences from the fuel handling accident; (3) the staff's independent analyses using the assumptions in Regulatory Guide 1.25, Sections C.1.a through C.1.k, and (4) the Watts Bar Technical Specifications relating to fuel-handling and ventilation system operation.

SSER 4 Section 15.4.5, Fuel Handling Accident, on page 15-2 states:

The staff's conclusion is based on (1) the staff's determination that the design features and plant procedures at Watts Bar meet the requirements of GDC 51 with respect to radioactivity control; (2) the staff review of the applicant's assumptions and analyses of the radiological consequences from the fuel-handling accident, (3) the staff's independent analyses using the assumptions in RG 1.25, Sections C.1.a through C.1.k, and (4) the Watts Bar Technical Specifications relating to fuel-handling and ventilation system operations.

TS Section: 3.9 and 3.7.12
TRM Section: 3.9

Section 3.9 or the TS and TRM provide requirements for fuel handling. TS section 3.7.12 places requirements on the Auxiliary Building Gas Treatment System for fuel handling operations.

This item is CONSISTENT.

51. SER Appendix C, Nuclear Regulatory Commission Unresolved Safety Issues, in section C.5, Discussion of Tasks as The Relate to Watts Bar Unit 1 and 2, states for issue A-3, Westinghouse Steam Generator Tube Integrity (page C-8 and C-9):

Specific measures such as steam generator design features and a secondary water chemistry control and monitoring program, that the applicant has employed to minimize the onset of steam generator tube problems are described in Section of this SER. In addition, Section 5.4.2 discusses the inservice inspection requirements. As described in Section 10.2, the applicant has met all current requirements regarding steam generator tube integrity. The Technical Specifications will include requirements for actions to be taken in the event that steam generator tube leakage occurs during plant operation.

TS Section: 3.4.13

The Action statements of LCO 3.4.13 on page 3.4-30 contain requirements for steam generator tube leakage.

This item is CONSISTENT.

52. SER Appendix C, Nuclear Regulatory Commission Unresolved Safety Issues, in section C.5, Discussion of Tasks as The Relate to Watts Bar Unit 1 and 2, states for issue A-11 Reactor Vessel Materials Toughness (page C-9 and C-10):

Resistance to brittle fracture, a rapidly propagating catastrophic failure mode for a component containing flaws, is described quantitatively by a material property generally denoted as "fracture toughness." Fracture toughness has different values and characteristics depending upon the material being considered. For steels used in a nuclear reactor pressure vessel, three considerations are important; first, fracture toughness increases with increasing temperature; second, fracture toughness decreases with increasing load rates; and third, fracture toughness decreases with neutron irradiation. In recognition of these

(2) Automatic Transfer of Loads Between Preferred Offsite Circuits

In SSER 2, the staff stated that the testing requirement for the automatic transfer of loads and the design which prevents a faulted or overloaded bus from being automatically transferred was to be reviewed with the Technical Specifications. With respect to the automatic transfer from the first preferred power source to the second alternate preferred power source and automatic transfer of non-faulted buses, the applicant committed by letter dated April 17, 1985, to rack out the second alternate power circuit breaker on each of four Class 1E 6.9-kV buses. Thus, the review of testing requirements was no longer necessary.

Subsequently, the design was changed to utilize automatic transfers of Class 1E and other buses between offsite sources as described in other sections of this report. The applicant will rely on coordination of protective devices to ensure that faulted or overloaded buses are not transferred from one source to the other. The staff considers this item to be acceptably resolved without technical specification requirements for surveillance of the automatic transfer/overloaded bus transfer prevention circuitry. The proposed requirement in SSER 2 is withdrawn.

TS section: None

SSER 13 deleted the requirement identified in SSER 2.

This item is CONSISTENT.

55. SSER 3 Section 15.3.2, Steamline Break, on page 15-1 states:

With regard to the proposed changes to the Technical Specifications and heat tracing for the BIT, the applicant stated that the current requirement was due to high boron concentration in the BIT and associated piping. Reduction of the boron concentration requirement to zero ppm would eliminated all Technical Specifications concerning BIT boron concentration, temperatures, and associated surveillance, including heat tracing, since heat tracing would only be required for boron concentrations above 4 weight percent corresponding to approximately 7,000 ppm.

TS Section: None

No requirements for BIT boron concentration or heat tracing were included in the final draft TS or TRM.

This item is CONSISTENT.

56. SSER 3 Section 15.3.6, Anticipated Transients Without Scram, on page 15-5 and 6 states:

(12) Verify that the response time of the automatic shunt trip feature will be tested periodically as shown to be less than or equal to than assumed in the FSAR analysis or that specified in the Technical Specifications.

The applicant notes that the response time of the automatic shunt trip will be tested and used in demonstrating that the reactor trip system instrumentation response times specified in the Technical Specifications are within their limits. The staff finds this acceptable.

(13) Propose Technical Specification changes to require periodic testing of the under voltage and shunt trip functions and the manual reactor trip switch contacts and wiring.

uncertainty in a four-loop plant. When using normalized elbow taps (one elbow tap per loop), the total uncertainty is given as. +/-2.1 percent. The comparable analysis submitted for the Watts Bar plant yields a value of +/-1.5 percent for the calorimetric uncertainty and total uncertainty of +/-1.7 percent, which includes the normalized elbow taps (one elbow tap per loop). A conference call was held between TVA, Westinghouse Electric Corporation, and NRC on February 5, 1985, to clarify the results of the analysis. It was found that several of the parameters that are used to arrive at the measurement uncertainty for the elbow tap flow readings (such as sensor temperature and pressure effects) have a negligible effect on accuracy. Therefore, the Watts Bar flow measurement uncertainties are less than those in the generic Westinghouse analysis. The staff noted that the Watts Bar flow measurement analysis values are identical to those of a similar analysis made for the William B. McGuire plant, and therefore accepts the flow measurement uncertainty value of +/-1.7%, which, with an additional 0.1% penalty required to account for venturi fouling brings the value to +/-1.8%. The staff is currently developing the Watts Bar Technical Specifications, and will report the final approved flow measurement uncertainty, if different from the value here, in a future SER supplement. This issue was tracked under TAC 63628.

TS Section: B 3.4.1 page B 3.4-2

The values for the penalty associated with venturi fouling are identified in the Bases are consistent with the SER.

This item is CONSISTENT.

59. SSER 5 Section 6.2.4, Containment Isolation System, on page 6-3 states:

In addition, Supplement No. 3 to the SER stated that operability of the containment purge/vent isolation valves during LOCA-induced pressure transients inside containment was still being reviewed by the staff. That review was completed and has been issued (letter from P. S. Tam, NRC, to 0. D. Kingsley, TVA, dated July 12, 1990). The staff concluded that the isolation valves can close against the buildup of pressure in containment in the event of a design-basis accident if the lower containment isolation valves are physically blocked to an opening angle of 50° or less. The requirement will be reflected in the Technical Specifications. Thus, proposed License Condition 8 is considered resolved.

TS Section: SR 3.6.3.7 page 3.6-13.

SR 3.6.3.7 verifies that each 24 inch containment lower compartment purge supply and exhaust isolation valve is blocked to restrict the valve from opening > 50° .

This item is CONSISTENT.

60. SSER 5 Section 6.4, Control Room Habitability, on page 6-4 states:

In an April 26, 1985, letter, the applicant proposed to remove the main control room air intake chlorine detectors. By letter dated May 15, 1985, TVA supplied additional information. The Technical Specifications will appropriately reflect this change. Sodium hypochlorite is used for water treatment instead of chlorine, and thus only a small quantity of chlorine will be stored on site for laboratory use. Negligible amounts of chlorine will be stored off site in close proximity to the plant. In addition, review of the number of chlorine shipments past the Watts Bar plant site indicates the number is sufficiently low that a toxic gas consequence analysis for chlorine shipment accidents need not be performed. The staff concludes that the removal of the chlorine detectors will not impair the safe operation of the plant.

62. SSER 6 Section 1.14.2, Generic Letters, on page 1-23 states:

Generic Letters 80-02 and 83-26, Quality Assurance Requirements Regarding Diesel Generator Fuel Oil

TVA Response NRC Action	Plant Technical Specifications are being developed, and when issued, will reflect the recommendations of this generic letter where appropriate. TVA addressed the issues in FSAR Section 9.5.4.2.
Implementation Status	Full implementation when Technical Specifications requirements are imposed.

TS Section: 3.8.3 and 5.7.2.16

This item is CONSISTENT.

63. SSER 6 Section 1.14.2, Generic Letters, on page 1-24 states:

Generic Letter 80-14, LWR Primary Coolant System Pressure Isolation Valves

TVA Response NRC Action	Guidance in this generic letter will be incorporated into the plant Technical Specifications, currently under development. TVA addressed this issue in FSAR 5.2.7.4.
Implementation Status	Full implementation when Technical Specifications requirements are imposed.

TS Section: 3.4.14

The TS requires Pressure Isolation Valves to be tested in accordance with the applicant's inservice test program.

This item is CONSISTENT.

64. SSER 6 Section 1.14.2, Generic Letters, on page 1-24 states:

Generic Letter 80-30, Clarification of the Term "Operable" as It Applies to Single Failure Criterion for Safety Systems Required by Technical Specifications

TVA Response NRC Action	Guidance being incorporated in the plant Technical Specifications.
Implementation Status	Open, pending issuance of the plant Technical Specifications.

TS Sections: 1.1, 3.0.3 and 3.05. Bases 3.03

The TS implementation of the items in the Generic Letter were not implemented exactly as specified by the Generic Letter, however, the appropriate sections were implemented consistent with NUREG-1431.

68. SSER 6 Section 1.14.2, Generic Letters, on page 1-27 states:

Generic Letter 84-15, Proposed Staff Actions To Improve and Maintain Diesel Generator Reliability

TVA Response	Plant Technical Specifications are being developed, and guidance in this generic letter will be considered.
NRC Action	
Implementation Status	Full implementation when Technical Specifications requirements are imposed.

TS Section: None

This item is CONSISTENT.

69. SSER 6 Section 1.14.2, Generic Letters, on page 1-28 states:

Generic Letter 85-19, Reporting Requirements on Primary Coolant Iodine Spikes

TVA Response	Guidance will be incorporated in the plant Technical Specifications or other appropriate document.
NRC Action	
Implementation Status	Full implementation when Technical Specifications requirements are imposed.

TS Section: 3.4.16

The requirement for shutdown if coolant iodine activity limits are exceeded for 800 hours in a 12-month period are not included in the TS submitted by the applicant.

This item is CONSISTENT.

70. SSER 6 Section 1.14.2, Generic Letters, on page 1-29 states:

Generic Letter 87-06, Leak Tight Integrity of Pressure Isolation Valves

TVA Response	Guidance will be incorporated in the plant Technical Specifications or other appropriate document.
NRC Action	
Implementation Status	Full implementation when Technical Specifications requirements are imposed.

TS Section: 3.4.14

TVA Response NRC Action	Not needed. Action is voluntary for operating reactor licensees. For Watts Bar, the guidance will be incorporated in the plant Technical Specifications.
Implementation Status	Not Applicable.

TRM Section: 3.7.3

Requirements for snubbers was located in the TRM instead of the TS as part of the adoption of the Improved Technical Specifications (ITS).

This item is CONSISTENT.

74. SSER 6 Section 1.14.2, Generic Letters, on page 1-32 states:

Generic Letter 91-01, Removal of the Schedule for Withdrawal of Reactor Vessel Material Specimens from Technical Specifications

TVA Response NRC Action	Not needed. Action is voluntary for operating reactor licensees. For Watts Bar, the guidance will be incorporated in the plant Technical Specifications.
Implementation Status	Not Applicable.

TS Section: None

Requirements for removal of Reactor Vessel Material Specimens is not contained in the TS submitted by the applicant.

This item is CONSISTENT.

75. SSER 6 Section 11.7.2, Primary Coolant Outside the Containment (TMI Item III.D.1.1), on page 11-1 states:

In SSER 5, the staff concluded that the applicant's leakage-reduction program is acceptable, and stated that proposed License Condition 24 will be resolved if the applicant accepts the inclusion of the waste gas disposal system (WGDS) in the leakage-reduction program.

By letter dated March 27, 1986, the applicant justified excluding the WGDS from the program (i.e., Section 6.8.5 of the then-proposed Watts Bar Technical Specifications). However, in Generic Letter (GL) 89-01, dated January 31, 1989, the staff requested that licensees and applicants relocate all radiological effluent technical specifications (RETS) to the respective plant offsite dose calculation manual (ODCM). Among line items to be relocated is the specification on the WGDS. Therefore, in accordance with the guidance of GL 89-01, the staff would exclude the WGDS from the Watts Bar Technical Specifications, but expects to see it included in the ODCM. The staff is reviewing the ODCM and is tracking it by, TAC 77553. Proposed License Condition 24 will be considered fully resolved when the ODCM is issued accordingly.

SSER 13 Section 15.2.4.1, Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from Zero Power Conditions, on page 15-2 states:

By letter dated November 5, 1993, the applicant proposed to modify LCO 3.4.6 and the associated "Bases" section to include a requirement that two reactor coolant pumps should be running whenever the rods are capable of withdrawal in Mode 4. The proposed modification of the LCO resolves the staff's concern, and conforms to the assumption used in SSER 7.

TS Section: 3.4.6 page 3.4-11 and B 3.4.6 page B 3.4-27.

Limiting Condition for Operation 3.4.6 requires that two reactor coolant pumps be running whenever the rods are capable of withdrawal in Mode 4. The Bases also discusses the requirement for two reactor coolant pumps to be operating.

This item is CONSISTENT.

77. SSER 8 Section 4.4.3.4, Reactor Coolant System Temperature Measurement, on page 4-2 states:

The Eagle-21 system employs an algorithm that automatically detects a defective hot-leg RTD input signal and eliminates that input from the calculation of "T_{hot} average." This is accomplished by incorporating a redundant sensor algorithm (RSA) into the hot-leg temperature signal processing. The RSA determines the validity of each input signal and automatically rejects a defective input. The typical tolerance bandwidth for automatic rejection is from 2° to 6° (F). The exact value will be determined during startup based on actual measurements of hot-leg temperature. Because of hot-leg streaming, the temperature varies in the cross-section of the hot legs. The RTDs at the three locations in each hot leg are processed to get a "T_{hot} average" temperature. The Eagle-21 system can add a bias to the averaging calculation, if needed, in order to compensate for the loss of one of the three RTD sensor inputs. The bias considers the past history of the previous hot-leg readings. The input bias that is used to compensate "T_{hot} average" upon loss of one narrow-range Thot signal is based on "T_{hot} average" with three valid RTD inputs. There is one bias value associated with each narrow-range Thot RTD input signal. The bias value for each RTD is calculated while all three RTDs are considered to be valid by subtracting the average of the remaining two RTDs from the "T_{hot} average" value for the loop. Then, if an RTD should fail, "T_{hot} average" for that loop is calculated by adding the bias value for the failed RTD to the average of the remaining two RTDs. If a single RTD does fail, the value of "T_{hot} average" would be calculated as described above and an "RTD failure" alarm and status light indicating "trouble" would be activated in the control room. The failed RTD would be replaced during a subsequent plant outage. If two or three hot-leg RTDs in the same loop fail, a dedicated alarm and annunciator would be activated indicating a failed channel. The Watts Bar Technical Specifications or similar documents will detail the action that must be taken for a failed overtemperature delta-t/overpower delta-T channel.

TS Section: 3.3.1 page 3.3-9.

Action W of LCO 3.3.1 provides action for failed overtemperature delta-t and overpower delta-t channels.

This item is CONSISTENT.

78. SSER 8 Section 4.4.3.4, Reactor Coolant System Temperature Measurement, on page 4-2 states:

81. SSER 9, Appendix V, Section 3.3.11, Accumulator Tank Level and Pressure, on page 11 states:

Regulatory Guide 1.97 recommends Category 2 instrumentation with ranges of zero to 750 psig and 10 percent to 90 percent volume. Watts Bar has Category 3 pressure instrumentation for these accumulators with a range of zero to 700 psig. This range exceeds the technical specification requirements that require a nitrogen blanket pressure of 632 psig. A pressure relief valve maintains pressure below 700 psig. We find the present zero to 700 psig range acceptable. This range will enable the operator to see that the accumulators have discharged.

TS Section: SR 3.5.1.3

The TS contain a range for nitrogen pressure from $\geq$ 620 psig to $\leq$ 655 psig instead of a fixed value of 632 psig. The conclusions of the SSER are consistent with the range specified in TS.

This item is CONSISTENT.

82. SSER 9 Appendix V, Section 3.3.11, Accumulator Tank Level and Pressure, on page 12 states:

The applicant identified a range of 75 percent to 82 percent for the Category 3 level instrumentation. This range monitors the level of the accumulators during normal operation. We note that the range of the level instrumentation coincides with the technical specification requirements to maintain between 7,632 gallons and 8,264 gallons in each accumulator.

TS Section: SR 3.5.1.2

The TS contain requirements for accumulator level from $\geq$ 7717 gallons to $\leq$ 8004 gallons. These values are more restrictive than the values used in the SSER, so the conclusions of the SSER are consistent.

This item is CONSERVATIVE.

83. SSER 9 Appendix V, Section 3.3.11, Accumulator Tank Level and Pressure, on page 12 states:

The accumulators contain borated water and a blanket of pressurized nitrogen gas. The accumulators are a passive system. They provide a fast acting, high flow rate injection. The flow is into the cold legs of the reactor coolant system during the injection phase of a large loss of coolant accident recovery. The accumulators are isolated from the reactor coolant system during normal reactor operation by two series-connected check valves. Each accumulator also has a motor-operated isolation valve. These valves are open before power operation. Once open, their power is removed. Technical Specification surveillance requirements assure the status of these isolation valves. Should the reactor coolant system pressure decrease below the accumulator pressure, the check valves open. This allows the blanket nitrogen gas pressure to force the borated water into the reactor coolant system. The mechanical operation of the swing-disk check valves is the only action required for this injection. No external power source, initiating signal, or post-accident monitoring instrumentation is necessary for the accumulators to perform their safety function. The operator must take deliberate actions to close the isolation valves. Post-accident isolation of the accumulators from the reactor coolant system is not a safety function. Once the accumulators have automatically discharged, they have no further post-accident safety function. The level instrumentation will read off-scale (low) following a discharge and accumulator discharge is verifiable by the pressure instrumentation. Therefore, we find the range of the level instrumentation acceptable.

The TS or TRM do not contain requirements for ATWS equipment. The improved TS of NUREG-1431 do not contain specifications for ATWS equipment.

This item is CONSISTENT with current guidance for inclusion of ATWS equipment in TS. Issue is currently under generic staff review.

85. SSER 10 Section 12.4, Design Features, on page 12-1 states:

The applicant has designed the facility in five radiation zones based on radiation source intensity and occupancy requirements. Zones IV and V correspond to areas of the plant that are high-radiation areas as defined in 10 CFR Part 20. The applicant has proposed to control access to these high-radiation areas by technical specifications. The staff is developing the Watts Bar Technical Specifications and will ensure that appropriate requirements are imposed consistent with current guidelines. The staff finds the applicant's program meets the provisions of 10 CFR 20.1601(c) and the acceptance criteria in SRP Section 12.3 and is, therefore, acceptable.

TS Section: 5.11 page 5.0-36.

The TS contain the requirements for control of High Radiation Areas.

This item is CONSISTENT.

86. SSER 11 Section 9.5.4, Emergency Diesel Engine Fuel Oil Storage and Transfer System

In Section 9.5.4.2 of SSER 9, the staff clarified the capacity of the 7-day fuel oil storage tank and equated the storage capacity of each tank as 68,000 gallons. In FSAR Amendment 69, TVA revised the capacity to 70, 260 for each diesel generator. The revised capacity still exceeds the amount needed for a 7-day supply and, therefore, does not affect the staff's evaluation and conclusions given in the SER or any of its supplements. The minimum number of gallons required for 7 days of operation will be specified in the plant Technical Specifications, which constitutes Appendix A of the operating license. This effort was tracked by TACs M82644 and M82645.

TS Section: B 3.8.3 page B 3.8-43.

The Bases discussion for SR 3.8.3.1 states that value given in TS is the amount required for seven day operation.

This item is CONSISTENT.

87. SSER 12 Section 10.3.1, Main Steam Supply System (Up to and Including the Main Steam Isolation Valves)

In Section 10.3.1 of the SER, the staff stated that the main steam isolation valves (MSIVs) would close on "high-high" containment pressure, or high steam flow coincident with low steam generator pressure or low reactor coolant average temperature. With the update to the Eagle-21 process protection system as described in FSAR Amendment 72, the closing signals for the MSIVs now consist of "high-high" containment pressure, low steamline pressure, and high steamline pressure rate (decreasing). The MSIVs will close within 5 seconds after receipt of any one of these three signals. These are the same three signals used for MSIV isolation at the Sequoyah nuclear plant and approved by the staff in the Sequoyah Technical

TS Section: 3.3.5

The clarification in SSER 14 resolves the issue identified in SSER 13. The setpoints and allowable values for load-shed and diesel start relays are listed in table 3.3.5-1 of the Technical Specifications.

This item is CONSISTENT.

89. SSER 13 Section 8.3.1.2, Low and/or Degraded Grid Voltage Condition, on page 8-9 states:

- Technical Specifications limits for maximum and minimum limits that will be in the Technical Specifications

During the site review of August 7 and 8, 1991, the applicant stated that the setpoint for the load-shed relays was 4860 volts with a 5-second time delay to close contacts on complete loss of voltage. The staff will require the applicant to include the setpoint in the Technical Specifications for the load-shed relays and similar minimum limits for the diesel-start relays. Maximum limits for the load-shed relays will be included in the Technical Specifications based on the results of the applicant's analysis. At 75-percent voltage, the load-shed relays should not actuate to strip loads. Thus, the Technical Specifications should require, for example, that the capability of the relays not to trip when subjected to a voltage of 75 percent for 30 seconds be periodically demonstrated.

These items have now been included in the draft Technical Specifications. Therefore, the staff's concerns are resolved.

SSER 14 Section 8.3.1.2, Low and/or Degraded Grid Voltage Condition, on page 8-2 states:

The material that follows revises the discussion in SSER 13.

(1) Allowable Technical Specification Limits for the Inverse Time Delay Relay

In SSER 13, the staff stated that Technical Specifications should require, for example, that the capability of the relays not to trip when subjected to a voltage of 75 percent for 30 seconds be demonstrated. The staff implied that this had been included in the draft Technical Specifications. This statement was wrong. Instead, the staff required that the setpoints and allowable values for the load-shed and diesel start relays be included in the plant's Technical Specifications to resolve the concerns.

TS Section: 3.3.5 page 3.3-51.

SSER 14 deleted the requirement to test the capability of the relays not to trip when subjected to a voltage of 75 percent for 30 seconds. The remaining tests were incorporated in TS.

This item is CONSISTENT.

90. SSER 13 Section 8.3.1.7, Possible Interconnection Between Redundant Divisions Through the Normal and Alternate Power to the Battery Charger, on page 8-10 states:

In the SER, the staff looked for possible interconnections for components that can be transferred between redundant divisions. For each component that can be transferred, the applicant documented the commitment to keep the alternate feed breaker at the 480-volt shutdown boards open, to install an alarm in

92. SSER 13 Section 8.3.1.12, The Capability and Independence of Offsite and Onsite Sources when Paralleled During Testing, on page 8-14 states:

In a letter dated August 5, 1993, the applicant reiterated what was stated above and submitted specific details pertaining to how an EDG being tested in parallel with offsite power would respond during LOOP and LOOP-with LOCA scenarios. As stated above, during a LOOP event, the diesel generator's's output breaker would trip on overcurrent. The instantaneous overcurrent relay would reset automatically and as soon as load shedding was completed, the EDG's output breaker would close since the diesel generator's is already running at rated speed. For a LOOP concurrent with a LOCA, the same scenario applies (with a different load sequence) with the EDG loading being initiated before the design-basis item (10 seconds).

Subsequently, the applicant stated in a phone call initiated per 10 CFR 50.9 that this information was wrong. Further, the applicant stated that it was considering a design change. In a letter dated February 7, 1994, the applicant discussed the design change which modifies the EDG control circuitry so that the output breaker for the EDG that is being tested parallel with the grid will trip on a safety injection signal. This, in turn, will realign the EDG to its emergency mode and override the test controls. For a LOOP condition, the EDG's output breaker will trip on a fault signal associated with the normal offsite power which, in turn, de-energizes the associated 6.9-kV shutdown board leading to the load-shedding sequence.

TS Section: SR 3.8.1.17 page 3.8-14.

SR 3.8.1.17 tests the response of the diesel generator to an ESF signal while operating in parallel with the grid.

This item is CONSISTENT.

93. SSER 13 Section 8.3.1.13, The Use of an Idle Start Switch for Diesel Generators, on page 8-15 states:

FSAR Figure 8.3.25 provides the design details for the idle start switch. Also, the information was incorporated into the FSAR by Amendment 71. The applicant stated that the circuitry for bypassing the idle start switch on an accident signal would be periodically tested per the Technical Specifications as part of EDG testing. Therefore, this concern is resolved.

TS Section: SR 3.8.1.20

This surveillance verifies the idle start circuitry is disabled on any automatic or emergency start signal.

This item is CONSISTENT.

94. SSER 13 Section 8.3.2.5, Non-Safety Loads Powered from the DC Distribution System and Vital Inverters, on page 8-17 states:

(3) Describe how the battery capacity with margin will be maintained and periodically verified over the life of the plant.

During the site review of August 7 and 9, 1991, and subsequently in a letter dated September 13, 1991, the applicant stated that battery performance would be verified periodically by test. On the basis of this response, this item is resolved.

TS Sections: 3.8.4 and 3.8.5

A note is provided with each LCO listed above that states that Vital Battery V may be substituted for any of the required vital batteries. While no specific requirements exist concerning operability of Vital Battery V, it cannot be substituted unless it meets all operability requirements.

This item is CONSISTENT.

97. SSER 13 Section 8.3.3.5.2, Compliance with Regulatory Guides 1.108 and 1.118, on page 8-28 states:

In a letter dated February 7, 1994, the applicant stated that the FSAR would be revised to describe compliance with RG 1.9, Revision 3, "Selection, Design, Qualification, Testing, and Reliability of Diesel Generator Units Used As Onsite Electrical Power Systems at Nuclear Power Plants." As part of the FSAR revision, the applicant is deleting compliance statements for RG 1.108 which has been withdrawn by the staff. Since this, in effect, negates the staff's previous review conclusions pertaining to RG 1.1.08 ant its guidance for diesel generator testing, the applicants's new compliance with RG 1.9, Revision 3 (and an exception thereto), is considered an open item. The staff has told the applicant about this concern in a letter dated March 28, 1994. The staff will track resolution of this issue by TACs M89190 and M89110, and will convey its evaluation in an SER supplement.

SSER 15 Section 8.3.3.5.1, Compliance With Regulatory Guides 1.108 and 1.118, on page 8-3 and 8-4 states:

The material that follows replaces the entire discussion under Subsection (1) in SSER 14 and supplements Subsection (3) in SSER 14. Also, the material does not change the staff's original conclusions.

(1) Class IE Standby Power System Testing

In Section 8.1.5.3 of the FSAR, as updated by Amendment 86, the applicant indicated that the Watts Bar electrical system design does not fully comply with Position C.2.2.6 of RG 1.9 (Revision 3). Position C.2.2.6 of RG 1.9 (Revision 3) recommends that a combined safety injection actuation signal (SIAS) and loss of offsite power (LOOP) test be performed (as part of pre-operational and periodic testing programs) to demonstrate that the emergency diesel generator can satisfactorily respond to a LOOP in conjunction with a SIAS in whatever sequence they might occur (e.g., loss-of-coolant accident (LOCA) followed by delayed LOOP or LOOP followed by LOCA). In clarification, the applicant stated:

> The design basis at WBN is a simultaneous LOOP/LOCA, not LOOP followed by LOCA. Although there are some design features to meet the effects of LOOP followed by LOCA, there is no analysis to demonstrate the design will meet the DG voltage and frequency requirements.

On the basis of this clarification, the staff understood that an actual simulated LOOP followed by a LOCA test would not be performed. In place of an actual simulation, the staff understood that the operability of the additional design features installed to meet the effect of LOOF before a LOCA would be demonstrated by a number of different tests.

The following features will be tested as required by the Watts Bar Technical Specifications:

- Automatically sequenced accident loads remain below the continuous rating of the diesel.

- Use of manual actuation in accordance with plant emergency operating procedures could result in loading to 101 percent of the diesel generator's continuous rating.

- Loading of non-safety loads during an accident are limited to the short time rating of the diesel generator.

- The diesel generator's load rejection capability would be demonstrated as part of preoperational and periodic testing by using a load equal to between 90 and 100 percent of the diesel generator's continuous rating.

Criteria III and XI of 10 CFR Part 50, Appendix B, require that (1) measures be provided for verifying or checking the adequacy of design by design reviews, by the use of alternative or simplified calculational methods, or by the performance of a suitable testing program and (2) a test program be established to ensure that systems and components perform satisfactorily and that the test program include operational tests during nuclear power plant operation.

The staff concludes that the capacity and capability of the diesel generators to reject a complete loss of load without tripping, mechanical damage, or harmful over stresses will be demonstrated in accordance with the guidelines of Regulatory Guide 1.9 (Revision 3) as part of the preoperational and periodic test programs. The design, therefore, meets the requirements defined above of Criteria III and XI of 10 CFR Part 50, Appendix B, and is acceptable.

TS Section: 3.8.1 and 3.8.2

Surveillance requirements of section 3.8.1 incorporate the requirements for the tests identified in the SSERs. However, SR 3.8.19 on page 3.8-9 only tests that the diesel generator will be able to reject a load greater than or equal to it single largest post-accident load whereas the SSER discusses the ability to reject a load equal to between 90 and 100 of the diesels instantaneous rating. SR 3.8.19 is consistent with the improved TS of NUREG-1431.

This item is CONSISTENT.

98. SSER 14, Section 3.9.6.2, Valve Testing Program, on page 3-17 and 3-18 states:

During refueling outages maintenance is performed, as required, in an attempt to restore all CIVs to below their reference leak rates and as close to zero leakage as is reasonably achievable. This ensures the ability of the containment system to satisfy the integrated leak rate testing criteria and to provide adequate margin for valve degradation over the next fuel cycle. While every attempt is made to maintain CIVs at zero leakage or below their reference leak rates at all times, a valve leaking in excess of its reference value may remain operable and left "as is," provided that an evaluation finds it acceptable with 10 CFR 50 Appendix J. An example of such a situation would be a valve found to be leaking in excess of its reference leak rate in mid-fuel cycle, and for which all reasonable on-line maintenance efforts have been made. Such evaluation shall be based upon consideration of the effects on overall containment leakage and possible effects on adjacent piping and components, as well as consideration of time, cost, unit operations, and radio logical exposure required for corrective measures. While the maximum permissible leak rate at this time would, by plant Technical Specifications, be limited to the current margin between overall containment leakage and 0.6 L_a maximum single leakage is at all times administratively limited to a value

In the original Safety Evaluation Report (SER, 1982), the staff stated that a feedwater isolation signal was initiated by a high-high steam generator level, an engineered safety feature (ESF) (safety injection) actuation signal, or a reactor trip. In Amendment 82, the applicant revised Final Safety Analysis Report (FSAR) Section 10.4.7 to identify a new feedwater isolation signal and to clarify the isolation signal generated by a reactor trip. The applicant noted that a high-flood-level detection in either the south or north main steam valve (MSV) vault rooms also generates a main feedwater isolation signal. In addition, the applicant clarified that the main feedwater isolation would only be generated by a reactor trip if the reactor trip is coincident with a low reactor coolant average temperature (low Tavg) signal.

The new feedwater isolation signal initiated by high-flood-level detection in the MSV vault rooms was added to prevent the submergence of equipment governed by 10 CFR 50.49 (environmental qualification) located in the MSV vault room in the event of a double-ended main feedwater line break. The flood detectors consist of three safety-grade-level switches in each room with a 2-out-of-3 logic to provide channelized inputs to trains A and B for feedwater isolation. The staff is revising this section of the SER to make the description of feed-water isolation signals consistent with actual plant design. This revision does not affect the conclusions reached in Section 10.4.7 of the original SER.

During its review of FSAR Amendment 82, the staff noted an unrelated error in the SER. In the SER, the staff stated that the main feedwater regulation valves will close within 5 seconds of receipt of a feedwater isolation signal and that the main feedwater isolation valves will close within 6.5 seconds of receipt of the isolation signal. According to the FSAR, both the feedwater regulation valves and feedwater isolation valves will close within 6.5 seconds of initiation of the feedwater isolation signal. The staff could not determine the actual origin of this discrepancy, but assumes that it was probably related to actual valve stroke times versus time to close after generation of a feedwater isolation signal. The accident and containment analyses are based on a closure time 6.5 seconds from the initiation of the feedwater isolation signal. Therefore, the staff concludes that 6.5 seconds is acceptable for both valves and considers this a matter of clarification of the original SER. The conclusions reached in Section 10.4.7 of the original SER are, therefore, still valid.

The staff's efforts were tracked by TACs M88694 and M88695.

TS Section: 3.3.2 and SR 3.7.3.1
TRM Section: 3.3.2

TS section 3.3.2 reflects the addition of the MSV Vault Room Water Level isolations. SR 3.7.3.1 requires that the closure times for each MFIV, MFRV, and associated bypass valve is $\leq$ 6.5 seconds which is inconsistent with the value given in the FSAR. TRM Table 3.3.2.1 provides a response time limit of $\leq$ 8 seconds for feedwater isolation which is nonconservative to the value given in FSAR for closure times from the receipt of an initiation signal.

This item is DIFFERENT.

RESOLUTION:

The applicant is to revise the FSAR to reflect the correct response time.

101. SSER 15 Section 6.2.1.1, Containment Structure, on page 6-2 and 6-3 states: