Tennessee Valley Authority, Post Office Box 2000, Spring City, Tennessee 37381-2000

August 3, 2007

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop: OWFN P1-35
Washington, D.C. 20555-0001

Gentlemen:

In the Matter of) Docket No. 50-391
Tennessee Valley Authority)

WATTS BAR NUCLEAR PLANT (WBN) - UNIT 2 - REACTIVATION OF CONSTRUCTION ACTIVITIES

The purpose of this letter is to inform the Nuclear Regulatory Commission (NRC) Staff of TVA's intention to reactivate and complete construction activities at WBN Unit 2. As set forth in the TVA's Key Assumptions letter (Reference 1) and recognized by the Commission in the NRC Staff Requirements Memorandum (Reference 2), TVA will complete the project under the existing construction permit and request an Operating License (OL) pursuant to 10 CFR Part 50.

As background, on October 4, 1976, TVA submitted a dual-unit WBN OL application for both WBN Unit 1 and Unit 2 (Reference 3). WBN Unit 1 received a full power OL on February 7, 1996. WBN Unit 2, which currently is in deferred status, would be operationally the same at startup as WBN Unit 1. Therefore, TVA believes that, from regulatory, safety and plant operational perspectives, significant benefit is gained from aligning the licensing and design bases of WBN Units 1 and 2 to the fullest extent practicable. The Commission recognized these benefits in Reference 2.

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In furtherance of this objective, TVA will complete WBN Unit 2 in compliance with applicable regulations promulgated prior to and after the issuance of the WBN Unit 1 OL. In addition, the WBN Unit 2 licensing and design bases will incorporate modifications made to WBN Unit 1, and those modifications currently captured in the WBN Unit 1 five-year plan. This alignment of the WBN Unit 1 and 2 licensing and design bases will ensure that there is operational fidelity between the units and at the same time demonstrate and ensure that WBN Unit 2 complies with applicable NRC regulatory requirements.

In combination, the enclosure to this letter and the accompanying attachments provide the information to be submitted by a licensee at least 120 days prior to the reactivation of construction in accordance with Generic Letter 87-15, "Commission Policy Statement on Deferred Plants" (52 FR 38077, October 14, 1987). Some of the key points discussed in the enclosure include the following:

TVA currently plans to resume unrestricted construction activities in support of completion of WBN Unit 2 on December 3, 2007. TVA expects to complete construction and request an operating license prior to April 1, 2012.

TVA will submit a regulatory framework document for Watts Bar Unit 2 similar to the Browns Ferry Status of Unit 1 Restart Issues letters. This regulatory framework document will reference key correspondence, describe the open issues or commitments that need to be resolved to obtain an OL for WBN Unit 2, discuss the background of the issues or commitments, and describe their completion or status, as appropriate. This document will be periodically updated.

TVA anticipates making no changes to the Site Security Plan or the Site Emergency Plan for purposes of WBN Unit 2 construction reactivation. Should any changes to the Site Security Plan or the Site Emergency Plan be necessary they will be evaluated and submitted to NRC as required by applicable regulations.

A Unit Separation Program will ensure that WBN Unit 2 construction activities do not adversely affect the continued safe operation of WBN Unit 1. As part of this program, unit separation isolation boundaries will be administratively controlled using design output (i.e., engineering drawings) or the Equipment Clearance Procedure. Feasibility studies are underway to evaluate the possible creation of construction openings into WBN Unit 2 containment (both equipment and personnel access). The Unit Separation Program will require moving the secondary containment boundary back to the Auxiliary Building wall and removing the WBN Unit 2 Reactor Building from the Vital Area as well as the Radiological Controlled Area. This will allow construction access to the WBN Unit 2 containment and minimize traffic in the operating unit.

Prior to resuming construction activities on quality or safety-related structures, systems or components, the Quality Assurance program and procedures will be put in place.

TVA will work with the NRC staff to review any exemptions, reliefs and other actions which were specifically granted for WBN Unit 1 to determine whether the same allowance is appropriate for WBN Unit 2.

TVA's successful operation of WBN Unit 1 provides reasonable assurance that WBN Unit 2 also can be completed successfully and then started and operated in a safe and reliable manner. If you have any questions, please contact me at (423) 751-6016 or Masoud Bajestani, WBN Unit 2 Vice-President, at (423) 365-2351.

Sincerely,

William R. McCollum, Jr.
Chief Operating Officer

References:

1. TVA letter, Watts Bar Nuclear Plant (WBN) - Unit 2 - Key Assumptions for the Possible Completion of Construction Activities, dated April 3, 2007.
2. NRC Staff Requirements - SECY-07-0096 - Possible Reactivation of Construction and Licensing Activities for the Watts Bar Nuclear Plant Unit 2, dated July 25, 2007.

3. TVA letter, Final Safety Analysis Report for Operating License for Watts Bar, Units 1 and 2, dated October 4, 1976.

RRB:JEM:BQP
Enclosures

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Generic Letter 87-15, "Commission Policy Statement on Deferred Plants" establishes the NRC's position regarding quality assurance requirements, particularly the maintenance, preservation and documentation requirements for deferred plants, and establishes how new regulatory requirements will be applied to deferred plants upon reactivation. The Policy Statement calls for TVA to notify the NRC Staff of its intentions regarding the reactivation of Watts Bar Nuclear Plant (WBN) Unit 2 at least 120 days prior to the resumption of plant construction. The information requested in the generic letter (in italics) and TVA's responses are provided as follows:

Commission Policy Statement on Deferred Plants Request 1*: The proposed date for resuming construction, a schedule for completing construction, and a schedule for submittal of an operating license (OL) application, including a final safety analysis report (FSAR), if one has not already been submitted.*

TVA Response: TVA plans to resume limited construction activities on selected non safety-related equipment like Raw Cooling Water piping replacement and turbine-generator-condenser work on October 1, 2007. TVA also expects to initiate engineering walkdowns to fully determine the scope of work in late October 2007. TVA plans to resume unrestricted construction activities in support of completion of Watts Bar Unit 2 on December 3, 2007. TVA expects to complete construction and request an operating license prior to April 1, 2012.

The construction permit for WBN Unit 2 (CPPR-92) is valid through December 31, 2010. A timely application for the extension of the construction permit will be submitted in accordance with 10 CFR 2.109 and 10 CFR 50.55(b).

On October 4, 1976, TVA submitted a dual-unit WBN OL application that included the Final Safety Analysis Report (FSAR) (Amendment 23) for both Watts Bar Units 1 and 2. The FSAR for both WBN Units 1 and 2 was periodically updated as part of the OL review process. The FSAR for WBN Unit 2 was last updated (Amendment 91) on October 24, 1995, in preparation WBN Unit 1 fuel load and issuance of the low power license. There have been no substantive design changes to WBN Unit 2 since submittal of the last FSAR revision for WBN Unit 2 in 1995. The WBN Unit 1 FSAR has been updated in accordance with 10CFR 50.71e and includes the shared equipment supporting WBN Unit 1.

Approximately 18 months prior to requesting permission to load fuel, TVA plans to submit a revised OL application updating the FSAR for WBN Unit 2 and the Technical Specifications.

Commission Policy Statement on Deferred Plants Request 2*: The current status of the plant site and operating equipment.*

TVA Response: WBN Unit 1 is currently operating and WBN Unit 2 is in deferred status. Structures and systems that are shared between Unit 1 and Unit 2 are in operation supporting Unit 1. The structures important to safety that are shared are the Auxiliary Building, Control Building, Diesel Generator Building and the intake pumping station. Shared safety-related systems include the essential raw cooling water, component cooling water, fire protection, spent fuel cooling, fuel oil storage tanks, preferred and emergency electric power, chemical and volume control, radioactive waste, emergency gas treatment system, and Control and Auxiliary Building ventilation systems.

TVA anticipates making no changes to the Site Security Plan or the Site Emergency Plan for purposes of WBN Unit 2 construction reactivation. Should any changes to the Site Security Plan or the Site Emergency Plan be necessary they will be evaluated and submitted to NRC as required by applicable regulations.

Under current TVA requirements, maintenance and layup of Unit 2 equipment is controlled by TVA Procedure TI-273 "Preventive Maintenance for Non-transferred Features". In 2004, NRC approved a revision to the quality assurance plan that allowed TVA to terminate layup preventive maintenance for selected equipment. This program has been continually refined and over time the scope of equipment requiring specific maintenance activities has been reduced based on economic considerations and actual preventive maintenance results. TVA will not take credit for equipment maintained under this reduced program. TVA will confirm that structures, systems and components can perform their intended functions by testing, refurbishment or replacement.

As required, components will be refurbished to bring WBN Unit 2 equipment up to current TVA maintenance program and environmental qualification requirements. This includes, for example, the replacement of seals, gaskets, and packing necessary as a result of the extended construction and layup period for WBN Unit 2. Where operating plant preventive maintenance procedures call for specific intervals of component refurbishment or inspection, these will be implemented, as appropriate, prior to startup in order to ensure that required maintenance is within appropriate periodicity at the time of initial operation.

Commission Policy Statement on Deferred Plants Request 3*: A description of how any conditions established by NRC during the deferral have been fulfilled.*

TVA Response: TVA suspended construction of WBN Unit 2 in 1985, placed the unit in a construction layup status, and formally deferred WBN Unit 2 in 2000. TVA has not identified any specific conditions established by NRC during the deferral period. NRC has conducted periodic inspections of the Layup and Preservation Program implemented at WBN Unit 2 and documented its findings in Inspection Reports. Issues raised by NRC inspections have been acted upon and resolved. Any open issues that are identified will be addressed in the regulatory framework document.

***Commission Policy Statement on Deferred Plants Request 4**: A list of licensing issues that were outstanding at the time of the deferral and a description of the resolution or proposed resolution of these issues.*

TVA Response: TVA reviewed the Safety Evaluation Report (SER) and its Supplements (SSERs) (NUREG-0847) related to the operation of Watts Bar Nuclear Plant Units 1 and 2 against the 1981 version of the Standard Review Plan (NUREG 0800) to develop a list of the remaining outstanding issues open for WBN Unit 2. Attachment 1 provides the results of this review. Specifically, Attachment 1 lists the outstanding issues, the confirmatory issues and license conditions and provides a reference to the subsequent SSER that resolved them for Unit 1 and 2, as appropriate. As a result of the review, TVA identified the following three outstanding issues for WBN Unit 2:

- Preservice Inspection Program,
- Pressure / Temperature Limits for Unit 2, and
- Essential Raw Cooling Water (ERCW) for two-unit operation.

Each of these issues is discussed individually below:

Preservice Inspection Program:

The WBN Unit 2 Preservice Inspection Program was last submitted to NRC on April 30, 1990. TVA will provide a revised program for NRC approval.

Pressure / Temperature Limits for Unit 2:

TVA will provide the Pressure Temperature Limits Report for WBN Unit 2 for NRC approval.

ERCW for two-unit operation:

The existing Essential Raw Cooling Water (ERCW) pumps were sized in 1974. In order to license WBN Unit 2, a two-unit preoperational flow balance test will be required. In their present conditions, the ERCW pumps do not provide adequate flow margin to meet the acceptance criteria of a two-unit flow balance test.

An engineering study was performed to determine the best alternative for meeting the design requirements of the ERCW system for two-unit operation. The alternatives are currently being reviewed. Appropriate measures will be taken to ensure the system is fully capable of meeting design requirements for two unit operation.

Attachment 1 also provides a listing of NRC Bulletins, Generic Letters and Three Mile Island Items and their status with respect to WBN Unit 2. The Watts Bar Nuclear Performance Plan Corrective Action Programs and Special Programs are listed and their status with respect to Watts Bar Unit 2. The final section of the matrix provides a listing of NRC Regulatory Guides and their applicability to WBN Unit 2.

By way of background, on September 17, 1985, the NRC sent a letter to TVA pursuant to 10 CFR 50.54(f), requesting that TVA submit information on its plans for correcting then-existing problems with the overall management of its nuclear program as well as plans for correcting Watts Bar specific issues. In response to that letter, TVA prepared a Corporate Nuclear Performance Plan and a site specific plan for resolution of the outstanding Corrective Action Programs (CAPs) and Special Programs (SPs) for WBN Units 1 and 2 entitled, "Watts Bar Nuclear Performance Plan" (WBNPP). NRC approval of these implementing actions, primarily for WBN Unit 1, is documented in the SER on the Tennessee Valley Authority: Watts Bar Nuclear Performance Plan (NUREG-1232, Volume 4) and NUREG-0847.

TVA has evaluated the CAPs and SPs to determine their applicability to WBN Unit 2. TVA will resolve the WBN Unit 2 CAPs and SPs consistent with NUREG-1232 (Volume 4), NUREG-0847 and applicable regulations. Attachment 2 to this enclosure provides a description of the CAPs and SPs, a summary of their resolution for WBN Unit 1 and a description of their proposed resolution for WBN Unit 2. If, during this process, TVA determines that it is necessary to modify the criteria otherwise specified in NUREG-1232, then it will submit such changes to the NRC for review and concurrence.

Attachment 3 provides a listing of NRC Generic Letters, Bulletins and TMI Action Items issued before December 31, 1994 and outstanding for Watts Bar Unit 2. TVA will resolve these in accordance with the outstanding commitments.

A listing of Open Construction Deficiencies Reports (CDRs) will be provided as part of the regulatory framework document. TVA intends to resolve these issues as part of the licensing process.

Commission Policy Statement on Deferred Plants Request 5*: A listing of any new regulatory requirements applicable to the plant since construction was deferred, together with a description of proposed plans for compliance or a commitment to submit such plans by a specific date.*

TVA Response: The Final Safety Analysis Report (FSAR) was updated for both units on October 24, 1995. NRC regulatory requirements from Title 10 of the Code of Federal Regulations Part 50 that have become effective since December 31, 1994 are listed in Attachment 4 to this Enclosure. Although not specifically required by the Commission Policy Statement, Attachment 4 also lists Generic

Letters and Bulletins applicable to WBN Unit 2. TVA intends to address these issues, as appropriate, as part of the licensing process.

Commission Policy Statement on Deferred Plants Request 6*: A description of the management and organization responsible for construction.*

TVA Response: TVA intends to rely on a contractor or contractor team to complete the engineering and construction for Watts Bar Unit 2 with oversight by TVA personnel. Contractor selection is in progress. Attachment 5 describes the TVA construction completion organization.

Commission Policy Statement on Deferred Plants Request 7*: A description of all substantive changes made to the plant design or site since the CP was issued (for those plants for which an OL application has not been submitted).*

TVA response: This item does not apply to WBN Unit 2, as TVA submitted the WBN Unit 2 OL application to the NRC on October 4, 1976. The application included the FSAR (Amendment 23) for WBN Units 1 and 2.

Commission Policy Statement on Deferred Plants Request 8*: Identification of any additional required information that is not available at the time of reactivation and a commitment to submit this information at a specific later date.*

TVA Response: In order to further address the issues discussed under the Commission Policy Statement for Deferred Plants Requests 4 and 5 above, TVA will submit a regulatory framework document for Watts Bar Unit 2 similar to the Browns Ferry Status of Unit 1 Restart Issues letters. This regulatory framework document will reference key correspondence, describe the open issues or commitments that need to be resolved to obtain an OL for WBN Unit 2, discuss the background of the issue, and describe their completion or status, as appropriate. TVA will provide this regulatory framework submittal for WBN Unit 2 by January 31, 2008.

Subsequent to the initial submittal, TVA will provide periodic updates, as appropriate, until the WBN Unit 2 commitments related to fuel load, startup and power operation are complete. These updates will provide formal notification of the completion of each NRC Bulletin, Generic Letter, Nuclear Performance Plan CAP and SP, and TMI Action Item for the applicable WBN Unit 2 plant milestone.

Commission Policy Statement on Deferred Plants Request 9*: As necessary, an amendment to the OL application (revised FSAR) and a discussion of the bases for all substantive site and design changes that have been made since the last FSAR revision was submitted (for those plants which were already under OL review at the time of deferral).*

TVA Response: TVA submitted the OL application for WBN Unit 2 to the NRC on October 4, 1976. The application included the FSAR (Amendment 23) for WBN Unit 1 and 2. The FSAR for WBN Unit 2 was last updated (Amendment 91) on October 24, 1995, in preparation for issuance of the WBN Unit 1 fuel load and low-power OL.

It is important to recognize that WBN Unit 2 was substantially complete when TVA suspended construction in 1985. There have been no substantive design changes issued since submittal of the last FSAR revision for WBN Unit 2 in 1995.

TVA plans to provide a red-line version of the WBN Unit 1 FSAR early in the project. The schedule for submitting this markup FSAR will be provided in the regulatory framework document.

Attachments:

Attachment 1 - Standard Review Plan / Safety Evaluation Report and Supplements – NUREG-0847 Review Matrix.

Attachment 2 - Outstanding Corrective Action Programs and Special Programs

Attachment 3 - Listing of Generic Letters, Bulletins and TMI Action Items issued before 1995

Attachment 4 – List of new Regulatory Requirements and Generic Communications

Attachment 5 – Construction Completion Organization

Attachment 6 – Listing of Commitments made in letter

Attachment 1 - Standard Review Plan / Safety Evaluation Report and Supplements – NUREG-0847 Review Matrix

Standard Review Plan / Safety Evaluation Report and Supplements (NUREG 0847) Review Matrix

Chapter 1 - Introduction and General Description of Plant
Chapter 2 - Sites Characteristics
Chapter 3 - Design of Structures, Components, Equipment, and Systems
Chapter 4 - Reactor
Chapter 5 - Reactor Coolant System and Connected Systems
Chapter 6 - Engineered Safety Features
Chapter 7 - Instrumentation and Controls
Chapter 8 - Electric Power
Chapter 9 - Auxiliary Systems
Chapter 10 - Steam and Power Conversion System
Chapter 11 - Radioactive Waste Management
Chapter 12 - Radiation Protection
Chapter 13 - Conduct of Operations
Chapter 14 - Initial Test Program
Chapter 15 - Accident Analysis
Chapter 16 - Technical Specifications
Chapter 17 - Quality Assurance
Chapter 18 – Human Factors Engineering (Reviewed in the SER as Control Room Design Review)
Chapter 19 – Severe Accidents

The NRC issued an OL Safety Evaluation Report (SER), NUREG-0847 for WBN1 and WBN2 in June 1982. The SER documented NRC's review of the WBN1 and WBN2 designs against Federal Regulations, construction permit criteria, and the NRC Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants ("SRP") otherwise known as NUREG-0800 (Revision 2 dated July 1981). Open issues raised by the reviewer in the SER that were not closed out when the SER was issued were classified into outstanding issues, confirmatory issues, and proposed license conditions. The staff listed 17 outstanding issues in the SER. Additional outstanding issues were added in Supplemental SERs (SSERs) for a total of 28. The SER listed 42 confirmatory actions; issue 43 was added in SSER6. There were 44 proposed Licensing conditions.

Attached is the matrix showing the NRC SRP sections (NUREG 0800) and the WBN1 and 2 SER (June 1982) and the WBN SSERs (1982-1995). The outstanding, confirmatory issues and license conditions are listed and a reference to the subsequent SSER that resolved them for Unit 1 and 2, as appropriate. Based on TVA's review of the SER and the SSERs, the open issues (outstanding issues, confirmatory issues, and proposed license conditions) for Watts Bar Unit 2 are shown in grey.

1982 SER	10 CFR / GDC Note 2	CURRENT SRP TITLE Note 1	ISSUES from 1982 SER	Later SSERS	Guidance (GL, Bulletins , other)	Regulatory Guides Notes 1, 2	Additional Information

Chapter 2 Site Characteristics

2.1.1	50.34(a)(1), 100.3, .10, *.11*	2.1.1 - Site Location and Description	No open issues				
2.1.2	*50.33, 50.34(a)(1)*, 100.3, .10, .11	2.1.2 - Exclusion Area Authority and Control	No open issues				
2.13-2.1.4	50.34(a)(1), 100.3, .10, .11	2.1.3 - Population Distribution	No open issues			1.70, 4.7	
2.21-2.2.3	*50.34(a)(1)(i), 100.10*	2.2.1- 2.2.2 - Identification of Potential Hazards in the Site Vicinity	No open issues			1.78, 1.91, 1.95	
2.2.2	50.34(a)(1), (a)(1)(ii); 100, Subpart A; 100.10, GDC 3, 4, 5	3.5.1.5 – Site Proximity Missiles (Except Aircraft) 3.5.1.6 - Aircraft Hazards	No open issues				
2.2.3	*50.34(a)(1)(i)*	2.2.3 - Evaluation of Potential Accidents	No open issues			1.70*R3*	
2.3.1	100.10(c)(2), GDC 2, 4	2.3.1- Regional Climatology	No open issues			1.76*R1*, 1.27	
2.3.2	100.10(c)(2), GDC 2	2.3.2 - Local Meteorology	No open issues			1.23, 1.70	
2.3.3	50.47(b)(4), (b)(8), (b)(9), Part 50 Section IV.E.2 of App E, App I; 100.10(c)(2), 100.11(a); *GDC 19*; Part 20, Subpart D	2.3.3 - Onsite Meteorological Measurements Programs	No open issues			1.23	
2.3.4	100.11(a), *GDC 19*	2.3.4 - Short term Dispersion Estimates for	No open issues			1.23, 1.78, 1.145, *1.194*,	

1982 SER	10 CFR / GDC Note 2	CURRENT SRP TITLE Note 1	ISSUES from 1982 SER	Later SSERS	Guidance (GL, Bulletins , other)	Regulatory Guides Notes 1, 2	Additional Information
		Accidental Releases				*1.3*, *1.4*, 1.5, 1.24, 1.25, 1.77, *1.98*, 1.70	
2.3.5	*20, Par D*	2.3.5 - Long term Atmospheric Dispersion Estimates for Routine Releases	No open issues			1.23, 1.109, 1.111, 1.112,1.70	
2.4.2	100.10(c), GDC 2	2.4.1 - Hydrologic Description	No open issues			*1.27*, *1.29*, *1.59*, *1.102*, 1.70	
2.4.3	50.34(a)(1), 100.3, .10, *.11*	2.1.1 - Site Location and Description	No open issues			*1.206*	
2.4.3	100, GDC 2	2.4.2 Floods	No open issues			*1.27*, 1.29, 1.59, 1.102, 1.70	
2.4.3	100.10(c), GDC 2	2.4.3 - Probable Maximum Flood (PMF) on Streams and Rivers	No open issues			*1.27*, 1.29, 1.59, 1.102, 1.70	
2.4.3	100.10(c), GDC 2, *44*	2.4.4 - Potential Dam Failures	No open issues			*1.27*, 1.29, 1.59, 1.102, 1.70	
NA	*100.10 (c), GDC 2, 44*	2.4.5 – Probable Maximum Surge and Seiche Flooding	Not addressed in SER			*1.27*, 1.29, 1.59, 1.102, 1.70	
NA	*100.10(c), GDC2*	2.4.6 - Probable Maximum Tsunami Hazards	Not addressed in SER			*1.27*, 1.29, 1.59, 1.102, 1.70	
NA	*100.10(c), GDC 2*	2.4.7 – Ice Effects	Not addressed in SER			1.27, 1.29, 1.59, 1.102,	

1982 SER	10 CFR / GDC Note 2	CURRENT SRP TITLE Note 1	ISSUES from 1982 SER	Later SSERS	Guidance (GL, Bulletins , other)	Regulatory Guides Notes 1, 2	Additional Information
						1.70	
NA	*100.10(c), GDC 1, 2, 44*	2.4.8 - Cooling Water Canals and Reservoirs	Not addressed in SER			*1.27*, 1.29, 1.59, 1.102, 1.125, 1.70	
NA	*100.10(c), GDC 2, 44*	2.4.9 – Channel Diversions	Not addressed in SER			*1.27*, 1.29, 1.59, 1.102	
2.4.3, 2.4.10	100.10(c); GDC 1, 2, 44	2.4.10 - Flooding Protection Requirements	No outstanding issues			1.29, 1.59, 1.102	
2.4.6	100.10(c); GDC 2, 44	2.4.11 - Low Water Conditions	No outstanding issues			1.27, *1.29*, 4.4	
2.4.7-2.4.8	50.55(a), 100.10(c), GDC 1, 2, 5, *44*	2.4.12 - Groundwater	2.4.8-Confirmatory issue for groundwater level for ERCW pipeline	Resolved SSER3		1.27	
2.4.9	100.10(c), *GDC 2*	2.4.13 - Accidental Releases of Liquid Effluents in Ground and Surface Waters	No open issues			1.113	
2.4.2-2.4.3	50.36, Part 100; 100.10(c); GDC 2 & 44	2.4.14 - Technical Specifications and Emergency Operation Requirements	No open issues			*1.29*, 1.59, 1.102	
2.5, 2.5.1	100.23, GDC 2	2.5.1 - Basic Geological and Seismic Information	No open issues			*1.165, 1.208*, 1.132, *1.198*, 4.7	
2.5	100.23, GDC 2	2.5.2 - Vibratory Ground Motion	No open issues			*1.165, 1.208*, 1.132, 1.60, *1.138*, 4.7	

1982 SER	10 CFR / GDC Note 2	CURRENT SRP TITLE Note 1	ISSUES from 1982 SER	Later SSERS	Guidance (GL, Bulletins , other)	Regulatory Guides Notes 1, 2	Additional Information
2.5	100.23, GDC 2	2.5.3 - Surface Faulting	No open issues			*1.165*, *1.208*, 1.132, *1.198*, 4.7	
2.5.4	50.55a, Part 50 Appendix B, 100.23, GDC 1, 2, 44	2.5.4 - Stability of Subsurface Materials and Foundations	2.5.4-Outstanding issue on liquefaction potential.	Resolved SSER3		1.27, 1.28, 1.132, 1.138, *1.198*	
			2.5.4-Confirmatory issue for analysis of sheetpile walls and material and geometric damping in soil-structure interaction analysis	Resolved SSER3			
			2.5.4-Confirmatory issue for design differential settlement of piping and electrical components	Resolved SSER3			
2.5.5	50.55a, Part 50 Appendix B, 100.23, GDC 1, 2, 44	2.5.5 - Stability of Slopes	No open issues			1.27, 1.28, 1.132, 1.138, *1.198*	

Chapter 3 - Design of Structures, Components, Equipment, and Systems

3.2.1	Part 50 *Appendix B*, Part 100 Appendix A, GDC *1*, 2, *61*	3.2.1 - Seismic Classification	3.2.1-Confirmatory Issue for ERCW upgrade to seismic category 1	Resolved SSER5		1.29, *1.143*, *1.151*, *1.189*	
			3.2.1- Confirmatory issue seismic classification of structures, systems, and	Resolved SSER5			

1982 SER	10 CFR / GDC Note 2	CURRENT SRP TITLE Note 1	ISSUES from 1982 SER	Later SSERS	Guidance (GL, Bulletins , other)	Regulatory Guides Notes 1, 2	Additional Information
			components important to safety				
3.2.2	*50.55a, Part 50 Appendix B*, GDC 1	3.2.2 - System Quality Group Classification	No open issues			1.26, 1.84, 1.85, *1.176, 1.201*	
3.3.1	GDC 2	3.3.1 - Wind Loadings	No open issues				
3.3.2	GDC 2	3.3.2 - Tornado Loadings	No open issues			*1.76R1*	
3.4	GDC 2, *4*	3.4.1 – Internal Flood Protection for Onsite Equipment Failures	No open issues			*1.29*	
NA	GDC 2	3.4.2 Analysis Procedures	Not addressed in SER				
3.5.1.1	GDC 4	3.5.1.1 - Internally Generated Missiles (Outside Containment)	No open issues			1.115	
3.5.1.2	GDC 4	3.5.1.2 - Internally Generated Missiles (Inside Containment)	No open issues				
3.5.1.3	GDC 4	3.5.1.3 - Turbine Missiles	No open issues			1.115, 1.117	
3.5.1.4	GDC 2, 4	3.5.1.4 - Missiles Generated by Tornadoes and Extreme Winds	No open issues			1.76	
3.5.1.4	50.34(a)(1)(ii), Part 100 Subpart A, 100.10, GDC 4	3.5.1.5 - Site Proximity Missiles (Except Aircraft)	No open issues			*1.117*, 1.91	
3.5.2	*100.11*, GDC 2, 4	3.5.2 - Structures, Systems, and	3.5.2-Confirmatory Issue modifications to protect	Resolved SSER2		1.13, 1.27, 1.115, 1.117	

1982 SER	10 CFR / GDC Note 2	CURRENT SRP TITLE Note 1	ISSUES from 1982 SER	Later SSERS	Guidance (GL, Bulletins , other)	Regulatory Guides Notes 1, 2	Additional Information
		Components to be protected from Externally Generated Missiles	Diesel Generators				
3.5.3	GDC 2 & 4	3.5.3 - Barrier Design Procedures	No open issues			1.76, 1.142	
3.6.1	GDC 2 & 4	3.6.1 - Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment	3.6.1-Outstanding issue involving main steam line break outside containment	Resolved SSER14			
3.6.2	*Part 100* Appendix A, GDC 4	3.6.2 - Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping	No open issues				
NA	*GDC 4*	3.6.3 - Leak-Before- Break Evaluation Procedures	Not addressed in SER	New section SRP 1987 - Reviewed in SSER5 - No outstanding issues		1.45	
3.7.1	Part 100, Subpart A & Appendix A, GDC 2	3.7.1 - Seismic Design Parameters	3.7.1.1-Outstanding issue involving update of FSAR for site specific spectra	Resolved SSER8		1.60,*1.165, 1.70, 1.208*	
3.7.2	Part 100, Subpart A & GDC 2	3.7.2 - Seismic System Analysis	3.7.2.1.2-Outstanding issue involving mass eccentricity	Resolved SSER8		1.60, *1.70, 1.92, 1.122, 1.132, 1.138*	
			3.7.2.12-Outstanding issue involving	Resolved SSER11			

1982 SER	10 CFR / GDC Note 2	CURRENT SRP TITLE Note 1	ISSUES from 1982 SER	Later SSERS	Guidance (GL, Bulletins , other)	Regulatory Guides Notes 1, 2	Additional Information
			comparison of Set A vs. Set B response				
3.7.3	Part 100, Subpart A & Appendix A, *100.23*, GDC 2	3.7.3 - Seismic Subsystem Analysis	3.7.3-Outstanding issue involving number of peak cycles to be used for OBE	Resolved SSER8		*1.70*	
			3.7.3-Outstanding issue involving use of code cases, damping factors for conduit and use of worst case, critical case and bounding case	Code case use, damping factors for conduit resolved SSER8, use of worst /critical / bounding case resolved for Unit 1 only in SSER12 (CAP/SP implementation issue resolved in IR 390/93-201)			
			3.7.3-Outstanding issue involving 1.2 multi-mode factor	Resolved SSER9			
3.7.4	*Part 20*	3.7.4 - Seismic Instrumentation	No open issues			1.12, *1.166*, *1.70*	
NA	*50.44*, 50.55(a), GDC 1,2,4,16, 50, *Appendix J*	3.8.1 Concrete Containment	Not addressed in SER			*1.7R2*, *1.35*, *1.35.1*, 1.70, 1.90, *1.91*, *1.115*, 1.136,	

1982 SER	10 CFR / GDC Note 2	CURRENT SRP TITLE Note 1	ISSUES from 1982 SER	Later SSERS	Guidance (GL, Bulletins , other)	Regulatory Guides Notes 1, 2	Additional Information
						1.84, 1.147	
3.8.1	*50.34(f)*, *44*, 55a; GDC 1, 4, 16, 50	3.8.2 - Steel Containment	3.8.1-Confirmatory issue to verify buckling methodology	Resolved SSER3		*1.7*, 1.57, 1.70, *1.84, 1.147, 1.193*	
			3.8-Outstanding issue involving load combinations and stress allowables	Resolved SSER9			
3.8.2	Part 50 *Appendix B*, 50.55a, *50.65*; GDC 1, 2, 4, 5, 50	3.8.3 - Concrete and Steel Internal Structures of Steel or Concrete Containments	No open issues			1.142, *1.199*, 1.57, *1.69*, *1.136*, 1.142, *1.143, 1.160, 1.199*	
3.8.3	Part 50 Appendix B, 50.55a, *50.65*; GDC 1, 2, 4, 5, 50	3.8.4 - Other Seismic Category 1 Structures	No open issues			1.142, 1.69, 1.70, 1.91, 1.115, *1.127*, 1.143, *1.160, 1.199*	
3.8.4	Part 50 Appendix B, 50.55a, 50.65; GDC 1, 2, 4, 5	3.8.5 - Foundations	No open issues			1.142, 1.70, *1.127, 1.142, 1.160*	
3.9.1	Part 50 Appendix B Section III, GDC 1, 2, 14, 15	3.9.1 - Special Topics for Mechanical Components	3.9.1-Outstanding issue involving assumption in piping analysis for water-hammer due to check valve slam	Resolved SSER13			
3.9.2.1, 3.9.2.2, 3.9.2.3 and	Part 50 Appendix B, 50.55a, GDC 1, 2, 4, 14 & 15	3.9.2 - Dynamic Testing and Analysis of Systems, Structures and Components	No open issues			1.20, *1.29*, 1.61, 1.68,1.92	

1982 SER	10 CFR / GDC Note 2	CURRENT SRP TITLE Note 1	ISSUES from 1982 SER	Later SSERS	Guidance (GL, Bulletins , other)	Regulatory Guides Notes 1, 2	Additional Information
3.9.2.4							
3.9.3.1, 3.9.3.2, 3.9.3.3 and 3.9.3.4	GDC 1, 2, *4*, 14 & 15	3.9.3 - ASME Code Class 1, 2 and 3 Components, Component Supports, and Core Support Structures	3.9.3.4-Outstanding issue, staff was awaiting TVA concurrence on their position with respect to margin for critical buckling of Reactor Coolant Pump support	Resolved SSER4	B 88-08, 88-11	*1.124, 1.130*	
			3.9.3.1-Outstanding issue involving use of experience data to qualify category I(L) piping	Resolved SSER8			
			3.9.3.3-Outstanding issue involving operating characteristics of main steam safety valves	Resolved SSER7			
			3.9.3.4-Confirmatory issue involving baseplate flexibility and its effect on anchor bolt loads	Resolved SSER8			
			3.9.3.4-Outstanding issue involving stiffness and deflection limits for seismic Category I pipe supports	Resolved SSER8			
			3.9.3.3-LC Relief and safety valve testing (II.D.1)	Resolved SSER3			
3.9.4	50.55a, GDC 1, 2, 14, 26, 27 & 29	3.9.4 - Control Rod Drive Systems	No open issues			1.26, 1.29	
3.9.5	50.55a, GDC 1, 2,	3.9.5 - Reactor Pressure	No open issues			*1.20*	

1982 SER	10 CFR / GDC Note 2	CURRENT SRP TITLE Note 1	ISSUES from 1982 SER	Later SSERS	Guidance (GL, Bulletins , other)	Regulatory Guides Notes 1, 2	Additional Information
	4 & 10	Vessel Internals					
3.9.6	*Part 50 Appendix B & J*, 50.55a, GDC *1, 2, 4, 14, 15*, 37, 40, 43, 46 & 54	3.9.6 – Functional Design, Qualification, and Inservice Testing of Pumps and Valves and Dynamic Restraints	3.9.6-Confirmatory issue-Required that Tech Spec include limiting condition for operation that requires plant shutdown or system isolation when leak limits are not met. Staff had not reviewed Tech Spec.	Resolved SSER14	GL 89-10, 90-09 & 96-05	*1.192, 1.174R1, 1.175, 1.148*	
			3.9.6 LC – Inservice testing of pumps and valves	Resolved SSER12			
NA		3.9.7 Risk – Informed Inservice Testing	Not addressed in SER		GL 89-04, 89-10 supplement 6	*1.174, 1.175*	
NA		3.9.8 Risk Informed Inservice Inspection of Piping	Not addressed in SER		GL 88-20, 89-08, 89-13 B 79-17, 88-01, 88-08, 88-11	*1.174, 1.178*	
3.10	Part 50 Appendix B, GDC 1, 2, 4, 14 & 30	3.10 - Seismic and Dynamic Qualification of Mechanical and Electrical Equipment	3.10-Generic outstanding issues involving adequacy of frequency test, peak broadening of response spectra, reconciling actual field mounting by welding vs. testing configuration mounted by bolting and *need for surveillance and*	Resolved all but adequacy of frequency test SSER6, resolved adequacy of frequency test SSER9		*1.100R2*, 1.61, 1.89, 1.92, *1.97R2,3,4*	

1982 SER	10 CFR / GDC Note 2	CURRENT SRP TITLE Note 1	ISSUES from 1982 SER	Later SSERS	Guidance (GL, Bulletins , other)	Regulatory Guides Notes 1, 2	Additional Information
			maintenance programs to address aging				
			3.10-Outstanding issue involving seismic classification of cable tray and conduits	Resolved SSER8			
3.11	Part 50 Appendix B, *50.49*, *50.67*, GDC 1, 2, 4, *23*, *NUREG-0737 II.B.2*	3.11 - Environmental Qualification of Mechanical and Electrical Equipment	3.11-Outstanding issue- TVA program not submitted at time of SER	Resolved SSER15			
NA	*GDC 1, 2, 4, 14, 15*	3.12 – ASME Code Class 1, 2 and 3 Piping Systems, Piping Components and their associated supports	Not addressed in SER		B 88-08, 88-11	*1.124, 1.130, 1.60, 1.84, 1.92, 1.122, 1.207*	
NA	*50.55(a), Part 50 Appendix A, B and G*	3.13 Threaded Fasteners – ASME Code Class 1, 2 and 3	Not addressed in SER		GL 91-17	*1.37, 1.65, 1.84*	

Chapter 4 - Reactor

4.2.1, 4.2.2, 4.2.3, 4.2.4	Part 50 Appendix K, *50.34*, 50.46, *50.67*, GDC 2, 10, 27, 35	4.2 - Fuel System Design	4.2.2- Confirmatory issue on thermal performance analysis code.	Resolved SSER2		1.4, 1.25, *1.157*, 1.126, 1.60, 1.77, *1.183*, *1.195*,	

1982 SER	10 CFR / GDC Note 2	CURRENT SRP TITLE Note 1	ISSUES from 1982 SER	Later SSERS	Guidance (GL, Bulletins , other)	Regulatory Guides Notes 1, 2	Additional Information
						1.196	
			4.2.3-Confirmatory issue on cladding-collapse calculations	Resolved SSER2			
			4.2.3 Confirmatory issue on Tech Spec to identify margins and to offset reduction in DNBR due to fuel rod bowing & incorporating residual bow penalty into the Tech Spec	Resolved SSER2			
4.3.1, 4.3.2, 4.3.3	GDC 10, 11, 12, 13, 20, 25, 26, 27, 28	4.3 - Nuclear Design	No open issues			*1.126*, 1.77	
4.4.1, 4.4.2, 4.4.3, 4.4.4, 4.4.5, 4.4.6, 4.4.7, 4.4.8	GDC 10, *12* NUREG-0737 II.F.2,	4.4 - Thermal and Hydraulic Design	4.4.4-Confirmatory issue on margin reduction due to effects of rod bow on DNBR	Resolved SSER2	GL 82-28, 86-09, 88-17	1.68, 1.133*R1*	
			4.4.3-Outstanding issue - Removal of RTD bypass system	Resolved SSER8			
			4.4.5-Confirmatory issue on review of Loose Parts Monitoring System (LPMS) startup report and inclusion of limiting conditions for LPMS in Tech Spec	Resolved SSER3			

1982 SER	10 CFR / GDC Note 2	CURRENT SRP TITLE Note 1	ISSUES from 1982 SER	Later SSERS	Guidance (GL, Bulletins , other)	Regulatory Guides Notes 1, 2	Additional Information
			4.4.8 – LC Detectors for Inadequate core cooling (II.F.2)	Resolved SSER10			
			4.4.5 LC Loose part monitoring system	Resolved SSER5			
4.5.1	*50.55a, GDC 1, 14, 26*	4.5.1 - Control Rod Drive Structural Materials	No open issues			*1.44, 1.85*	
4.5.2	50.55a, GDC 1	4.5.2 - Reactor Internals and Core Support Structure Materials	No open issues		GL 97-01	1.31, 1.44, 1.84	
4.6	GDC *4*, 23, 25, 26, 27, 28, 29	4.6 - Functional Design of Control Rod Drive System	No open issues				

Chapter 5 – Reactor Coolant System and Connected Systems

5.2.1.1	10 CFR 50.55a GDC 1	5.2.1.1 - Compliance with Codes and Standards Rule, 10CFR50.55a	No open issues			1.26	
5.2.1.2	10 CFR 50.55a GDC 1	5.2.1.2 - Applicable Code Cases	No open issues			1.84, 1.147, *1.192*	
5.2.2	10 CFR 50.34 (f)(2)(x) 10 CFR 50.34 (f)(2)(xi) Appendix G GDC 15, *30*, 31	5.2.2 - Overpressure Protection	5.2.2-Outstanding issue on staff review of sensitivity study of required safety valve flow rate versus trip parameter	Resolved SSER2	GL 96-03, 90-06, 82-16	*1.84*, 1.26, 1.29	
5.2.3	10 CFR 50.55a GDC 1, 4, 14, 30,	5.2.3 - Reactor Coolant Pressure Boundary	No open issues		GL 97-01, NRC Order EA-03-	1.31, 1.34, 1.36, 1.37,	

1982 SER	10 CFR / GDC Note 2	CURRENT SRP TITLE Note 1	ISSUES from 1982 SER	Later SSERS	Guidance (GL, Bulletins , other)	Regulatory Guides Notes 1, 2	Additional Information
	31	Materials			009	1.43, 1.44, 1.50, 1.71, *1.84*	
5.2.4	10 CFR 50.55a GDC 32	5.2.4 - Reactor Coolant Pressure Boundary Inservice Inspection and Testing	5.2.4-Outstanding issue – Unit 2 PSI program submitted April 30, 1990 with a partial listing of relief requests. This item tracks the staff review.	Resolved for Unit 1 only SSERs 10, 12 and 16 (Key Assumptions letter)	GL 88-05, B 88-09	*1.147, 1.150*	
			LC – Inservice inspection program	Resolved SSER12			
5.4.5	GDC 2, 30	5.2.5 - Reactor Coolant Pressure Boundary Leakage Detection	No open issues			1.29, 1.45	
			LC – Installation of reactor coolant vents (II.B.1)	Resolved for Unit 1 only SSER5 (IR 390/84-37)			
5.3.1	10 CFR 50.55a, *50.60* Appendix G, H GDC 1, 4, 14, 30, 31, 32	5.3.1 - Reactor Vessel Materials	No open issues			1.31, 1.34, 1.37, 1.43, 1.44, 1.50, 1.65, *1.99, 1.161*	
5.3.2	10 CFR 50.55a, *50.60, 50.61*, Appendix G	5.3.2 - Pressure-Temperature Limits, Upper Shelf Energy and	5.3.2-Outstanding issue on P-T limits for Unit 2 not provided. Staff will review	Outstanding issue on P-T limits for Unit 2		1.99, *1.154, 1.161*	

1982 SER	10 CFR / GDC Note 2	CURRENT SRP TITLE Note 1	ISSUES from 1982 SER	Later SSERS	Guidance (GL, Bulletins , other)	Regulatory Guides Notes 1, 2	Additional Information
	GDC 1, 14, 31, 32	Pressurized Thermal Shock	as part of Unit 2 Tech Spec.	(Key Assumptions letter)			
5.3.3	10 CFR 50.55a, *50.60, 50.61* GDC 1, 4, 14, 30, 31, 32	5.3.3 - Reactor Vessel Integrity	5.3.3-Outstanding issue that staff will complete evaluation of Unit 2 after receipt of P-T limits	Outstanding issue on P-T limits for Unit 2 (Key Assumptions letter)		1.99*R2*	
5.4.1.1, 5.4.2.1, 5.4.2.1		5.4 – Reactor Coolant System Components and Subsystem Design	No open issues				
5.4.1.1	10 CFR 50.55a (a)(i) GDC 1, 4	5.4.1.1 - Pump Flywheel Integrity (PWR)	No open issues			1.14	
5.4.2.1	10 CFR 50.55a (c), (d), e, *Appendix G* GDC 1, *4*, 14, 15, 30, 31	5.4.2.1 - Steam Generator Materials	No open issues			*1.31, 1.34, 1.36,* 1.37, *1.43, 1.44, 1.50, 1.65, 1.71, 1.84*	
5.4.2.2	*10 CFR 50.55a (g), 50.36, 50.65* GDC 32	5.4.2.2 - Steam Generator Program	5.4.2.2-Outstanding issue that staff will evaluate TVAs' proposed resolution to concerns about flow induced vibrations in Model D-3 SGs pre-heat region and address in SER supplement	Resolved SSER4		*1.121*	

1982 SER	10 CFR / GDC Note 2	CURRENT SRP TITLE Note 1	ISSUES from 1982 SER	Later SSERS	Guidance (GL, Bulletins , other)	Regulatory Guides Notes 1, 2	Additional Information
5.4.3	GDC 2, *4*, 5, 19, 34 NUREG 0737 III.D.1.1	5.4.7 - Residual Heat Removal (RHR) System	5.4.3-Confirmatory issues to verify installation of an RHR flow alarm and proper function of dump valves when actuated manually	Resolved SSER5 for Unit 1 (IR 390/84-28)	GL 88-17, 89-04, 90-06, 92-02 B 88-08, 88-04, 86-01	1.29, *1.82*	
			5.4.3-Outstanding issue involving natural circulation test to demonstrate ability to cool down & depressurize the plant, and that boron mixing is sufficient under such circumstances; or, if necessary, other applicable tests before startup after first refueling	Resolved SSER10			
5.4.4	GDC 2, 4	5.4.11 - Pressurizer Relief Tank	No open issues			*1.26*, 1.29	
5.4.5	10 CFR 50.46a, (b), 50.49, 50.55a GDC 14, *17, 19, 34, 36*	5.4.12 - Reactor Coolant System High Point Vents	No open issues			1.100, 1.92	

Chapter 6 – Engineered Safety Features

1982 SER	10 CFR / GDC Note 2	CURRENT SRP TITLE Note 1	ISSUES from 1982 SER	Later SSERS	Guidance (GL, Bulletins , other)	Regulatory Guides Notes 1, 2	Additional Information
6.1.1	Part 50 Appendix B Criteria IX & XIII, 50.55a, GDC 1, *4*,	6.1.1 - Engineered Safety Features Materials	No open issues			1.31, 1.36, 1.37, 1.44, 1.50, *1.84*	

1982 SER	10 CFR / GDC Note 2	CURRENT SRP TITLE Note 1	ISSUES from 1982 SER	Later SSERS	Guidance (GL, Bulletins , other)	Regulatory Guides Notes 1, 2	Additional Information
	14, 31, 35, 41						
6.1.2		6.1.2 - Protective Coating Systems (Paints) Organic Materials	No open issues			1.54	
6.2.1 contains 6.2.1.1 to 6.2.2.5	Part 50 Appendix K, 30.46, GDC 4,13, 16, 38, 39, 40, 50, 64	6.2.1 - Containment Functional Design	No open issues		GL 88-17	1.97, *1.157*, 1.4	
6.2.1.1	50.34(f)(3)(v)(A)1 & (B)1, GDC 13, 16, 38, 39, 40, 50, 64	6.2.1.1.B - Ice Condenser Containments	6.2.1.1-Confirmatory issue involves reviewing analysis that ensures that containment external pressure will not exceed design value of 2.0 psi	Resolved SSER3		1.97	
			LC – (6d) Accident monitoring instrumentation II.F.1 – containment pressure.	Resolved for Unit 1 only SSER5 (IR 390/84-59)			
			LC – (6e) Accident monitoring instrumentation II.F.1 – containment water level	Resolved for Unit 1 only SSER5 (IR 390/84-85)			
6.2.1.2	GDC 4 & 50	6.2.1.2 - Subcompartment Analysis	No open issues		SQN		
6.2.1.1.1	Part 50 Appendix K, GDC 50	6.2.1.3 - Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents (LOCAs)	No open issues				
6.2.1.1.1	GDC 50	6.2.1.4 - Mass and Energy	No open issues				

1982 SER	10 CFR / GDC Note 2	CURRENT SRP TITLE Note 1	ISSUES from 1982 SER	Later SSERS	Guidance (GL, Bulletins , other)	Regulatory Guides Notes 1, 2	Additional Information
		Release Analysis for Postulated Secondary System Pipe Ruptures					
6.2.1.3	Part 50 Appendix K I.D.2, 50.46(a)(1)(i) & (a)(1)(ii)	6.2.1.5 - Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Studies	No open issues			*1.157*	
6.2.2	50.46(b)(5), GDC 38, 39, 40	6.2.2 - Containment Heat Removal Systems	No open issues			*1.174R1*, 1.82	
6.2.3	Part 50 Appendix J, GDC 4, 16, 43	6.2.3 - Secondary Containment Functional Design	No open issues		SQN		
6.2.4	*Part 50 Appendix K, 50.34(f)(2)(xiv) & (xv), 50.63(a)(2),* GDC 1, 2, 4, 16, 54, 55, 56, 57 NUREG-0737 II.E.4.2 & II.E.4.4,	6.2.4 - Containment Isolation System	6.2.4-Confirmatory issue to install safety-grade isolation valves on 1" chemical feed lines joining feedwater lines to main steam line	Resolved SSER5	GL 83-02 & 88-17 SQN	1.11, 1.29	
			6.2.4-Outstanding issue for NRC to complete review of information provided by TVA to address Containment Purging During Normal Plant Operation	Resolved SSER3			
			6.2.4-Outstanding issue involving containment isolation using closed systems	Resolved SSER12			

1982 SER	10 CFR / GDC Note 2	CURRENT SRP TITLE Note 1	ISSUES from 1982 SER	Later SSERS	Guidance (GL, Bulletins , other)	Regulatory Guides Notes 1, 2	Additional Information
			LC – Modification of chemical feedlines	Resolved SSER5			
			LC – Containment isolation dependability	Resolved SSER5			
6.2.5	50.44, GDC 5, 41, 42, 43	6.2.5 - Combustible Gas Control in Containment	6.2.5-Outstanding issue for review of TVA provided additional information relative to discussion added to FSAR to address analysis of the production and accumulation of hydrogen within containment following onset of a LOCA	Resolved SSER4	SQN	1.7, 1.97, *1.155*	
			LC – (6f) Accident monitoring instrumentation II.D.1 – containment hydrogen	Resolved for Unit 1 only SSER5 (IR 390/84-85)			
			LC – 9 Hydrogen control measures	Resolved SSER8			
6.2.6	*Part 50 Appendix J, 100.10, 100.11, GDC 52, 53, 54*	6.2.6 - Containment Leakage Testing	No open issues			*1.163*	
6.2.7	50.55a, GDC 1, 16, 51	6.2.7 - Fracture Prevention of Containment Pressure Boundary	6.2.7-Confirmatory issue for TVA to confirm that the lowest temperatures which will be experienced by the limiting materials of the reactor containment pressure boundary under	Resolved SSER4		1.26	

1982 SER	10 CFR / GDC Note 2	CURRENT SRP TITLE Note 1	ISSUES from 1982 SER	Later SSERS	Guidance (GL, Bulletins , other)	Regulatory Guides Notes 1, 2	Additional Information
			the conditions cited by GDC 51 will be in compliance with the temperatures identified in the staff's analysis of fracture toughness requirements for load bearing component of the containment system				
6.3	Part 50 Appendix K, 50.46, *50.63*, GDC 2, 4, 5, 17, 27, 35, 36, 37 NUREG-0737 II.D.3, II.F.2, II.K.3.17, III.D.1.1,	6.3 - Emergency Core Cooling System	6.3.3-Confirmatory issue to provide a detailed survey of insulation material that could become debris post-LOCA	Resolved SSER2	B 79-24, 80-18, 86-03, 88-04, 88-08, GL 83-02, 88-17, 89-04 SQN	1.1, 1.29, 1.47, 1.52, 1.68, *1.155, 1.157, 1.82, 1.79*	
			6.3.1-Outstanding issue - involving removal of upper head injection system	Resolved SSER7			
			6.3.3-Outstanding issue involving containment sump screen design	Resolved SSER9			
6.4	GDC 4, 5, 19 NUREG-0737 III.D.3.4	6.4 - Control Room Habitability System	No open issues			*1.52R3, 1.78R1, 1.195, 1.196, 1.197, 1.183*	
6.5.1.1 to 6.5.1.4	100.11, GDC 19, 41, 42, 43, 61, 64	6.5.1 - ESF Atmosphere Cleanup Systems	No open issues		B 80-03	1.4, 1.7, 1.25, 1.52, *1.140, 1.183, 1.195*	
6.5.2	GDC 41, 42, 43	6.5.2 - Containment Spray as a Fission Product Cleanup System	No open issues			1.4, *1.183*	

1982 SER	10 CFR / GDC Note 2	CURRENT SRP TITLE Note 1	ISSUES from 1982 SER	Later SSERS	Guidance (GL, Bulletins , other)	Regulatory Guides Notes 1, 2	Additional Information
6.5.3	Part 100, GDC 41, 42, 43	6.5.3 - Fission Product Control Systems and Structures	No open issues			1.4, *1.52*	
6.5.4	GDC 41, 42, 43	6.5.4 - Ice Condenser as a Fission Product Cleanup System	No open issues				
6.6	50.55a, GDC 36, 37, 39, 40, 42, 43, 45, 46	6.6 - Inservice Inspection of Class 2 and 3 Components	6.6-Outstanding issue on additional information required on pre-service inspection program and identification of plant specific areas where ASME Code Section XI requirements cannot be met and supporting technical justification	Resolved for Unit 1only in SSER 10 (Key Assumptions letter)	GL 89-08		

Chapter 7 – Instrumentation and Controls

1982 SER	10 CFR / GDC Note 2	CURRENT SRP TITLE Note 1	ISSUES from 1982 SER	Later SSERS	Guidance (GL, Bulletins , other)	Regulatory Guides Notes 1, 2	Additional Information
7.1.1	50.55a(h), *50.67*	7.1 - Instrumentation and Controls - Introduction	7.1.3.1-Confirmatory issue to provide a list of all safety related functions and a summary of the setpoint analysis	Resolved SSER4	SQN	1.70, *1.152*	
7.2.1 to 7.2.6	50.55a(a)(1), a(h)(2), GDC *1*, 2, 4, 10, 13, *15, 19,* 20, 21, 22, 23, 24, 25, *29*	7.2 - Reactor Trip System	7.2.5-Confirmatory issue to address IEB 79-21 to alleviate temperature dependence problem associated with measuring	Resolved SSER2	SQN	1.22, 1.47, 1.53*R2*, 1.75*R2*, *1.105R3, 1.118, 1.152*	

1982 SER	10 CFR / GDC Note 2	CURRENT SRP TITLE Note 1	ISSUES from 1982 SER	Later SSERS	Guidance (GL, Bulletins , other)	Regulatory Guides Notes 1, 2	Additional Information
	NUREG-0737 I.D.3		SG water level				
7.3.1 to 7.3.6	50.55a(a)(1), a(h)(2), GDC *1*, 2, 4, *10, 13, 15, 16, 19*, 20, 21, 22, 23, 24, *29, 33*, 34, 35, 38, 41, *44* *NUREG-0737 I.D.3, II.E.1.2, II.E.4.2*	7.3 - Engineered Safety Features Systems	7.3.2-Confirmatory issue is commitment to make a design change to provide protection that prevents debris from entering level sensors	Resolved SSER2	SQN	1.22, 1.47, 1.53*R2*, 1.75*R2*, *1.105R3*, 1.118*R3*, *1.152*	
			7.3.5-Confirmatory issues to perform confirmatory tests to satisfy IEB 80-06 (to ensure that no device will change position solely due to reset action) and staff review of electrical schematics for modifications that ensure that valves remain in emergency mode after ESF reset	Resolved SSER3			
7.4.1 to 7.4.3	*50.55a(a)(1), a(h)(2), a(h)(3)*, GDC *1*, 2, 4, 13, 19, *24*, 34, 35, 38 *NUREG-0737 II.G.1*	7.4 - Safe Shutdown Systems	No open issues		SQN	*1.152R2, 1.189*	
7.5.1 to 7.5.4	*50.55a(a)(1), a(h)(2), a(h)(3)*, GDC *1*, 2, 4, 13, 19, *24* *NUREG-0737 I.D.3,*	7.5 - Information Systems Important to Safety	7.5.2-Outstanding issue involving RG 1.97 instruments following course of an accident	Resolved SSER9		1.97R1, 2,3,*4*, 1.47, *1.152R2*, 1.105*R3*, *1.7R3, 1.151*	

1982 SER	10 CFR / GDC Note 2	CURRENT SRP TITLE Note 1	ISSUES from 1982 SER	Later SSERS	Guidance (GL, Bulletins , other)	Regulatory Guides Notes 1, 2	Additional Information
	II.D.3, II.E.1.2, II.F.1, II.F.2, II.F.3, II.G.1						
7.6.1 to 7.6.9	*50.55a(a)(1), a(h)(2), a(h)(3), GDC 1*, 2, 4, 10, 13, 15, 16, 19, 24, 25, 28, 33, 34, 35, 38, 41, 44 *NUREG-0737 I.D.3*	7.6 - Interlock Systems Important to Safety	7.6.5-Confirmatory issue to install switches on the main control board for the operator to manually arm this system (overpressure protection provided by pressurizer PORVs)	Resolved SSER4		*1.47, 1.152R2*	
7.7.1 to 7.7.7	*50.55a(a)(1), a(h)2, a(h)(3), GDC 1, 10,* 13, *15*, 19, *24, 28, 29, 44*	7.7 - Control Systems	7.7.2 – LC – Status monitoring system, Bypassed and Inoperable Status Indication	Resolved SSER7		1.151, *1.152R2*	
NA	*50.62, 50.55a(h), GDC 1, 13, 19, 24*	7.8 Diverse Instrumentation and Control Systems	Not addressed in SER		GL 85-06	*1.152R2*	
NA	*50.55a(h), 50.62 50.34(f)(2)(v), GDC 2, 4, 13, 19, 21, 22, 23, 24, 29*	7.9 Data Communication Systems	Not addressed in SER			*1.152R2, 1.105R3, 1.22, 1.118, 1.53R1, 1.75R3, 1.180, 1.204*	
7.8	Appendix G GDC 15, 30, 31 NUREG-0737 II.D.1, II.D.3	5.2.2 Overpressure protection	7.8.1 LC - Confirm installation of acoustic monitoring system on Unit 2	Resolved for Unit 1 only SSER5 (IR 390/84-35)	GL 96-03, 90-06, 82-16	*1.84*, 1.26, 1.29	

Chapter 8 – Electrical Power Systems

1982 SER	10 CFR / GDC Note 2	CURRENT SRP TITLE Note 1	ISSUES from 1982 SER	Later SSERS	Guidance (GL, Bulletins , other)	Regulatory Guides Notes 1, 2	Additional Information
8.1		8.1 - Electrical Power - Introduction	No open issues				
8.2.1 to 8.2.4	*50.63, 50.65(a)(4),* GDC 2, 4, 5, 17, 18, 33, 34, 35, 38, 41, 44 *NUREG-0737 I.D.3,*	8.2 - Offsite Power System	8.2.2.1-Confirmatory issue to document additional information in FSAR on control power supplies and distribution system for the WB Hydraulic Plant Switchyard	Resolved SSER2	GL 2006-02	1.32, *1.155, 1.160R2, 1.182, 1.204,*	
			8.2.3-Outstanding issue involving compliance of design changes to the offsite power system with GDC 17 and 18.	Resolved SSER13			
8.3.1.1 to 8.3.1.9, 8.3.3.1 to 8.3.3.6	*50.55a, 50.63, 50.65(a)(4),* GDC 2, 4, 5, 17, 18, 33, 34, 35, 38, 44, 50 NUREG-0737 I.D.3, II.E.3.1, II.G.1	8.3.1 - AC Power Systems (Onsite)	8.3.1.1-Confirmatory issue to incorporate new design that provides dedicated transformer for each preferred offsite circuit in FSAR	Resolved SSER2	GL 1996-01, 2007-01 SQN	1.6, 1.9, 1.32, 1,47, 1.53, 1.63, 1.75, 1.81, 1.106, 1.118, *1.153, 1.155, 1.160, 1.182, 1.204*	
			8.3.1.2-Confirmatory issue to verify voltage drop analysis and testing	Resolved SSER13			
			8.3.1.6-Confirmatory issue to provide diesel generator reliability qualification test report	Resolved SSER7			

1982 SER	10 CFR / GDC Note 2	CURRENT SRP TITLE Note 1	ISSUES from 1982 SER	Later SSERS	Guidance (GL, Bulletins , other)	Regulatory Guides Notes 1, 2	Additional Information
			8.3.3.1.2-Confirmatory issue to verify design for bypass of thermal overload protective device	Resolved SSER2			
			8.3.3.2.3-Confirmatory issue for design of sharing raceway systems between units	Resolved SSER2			
			8.3.3.5.2-Confirmatory issue to incorporate commitment to test only one of four power trains of the plant at one time	Resolved SSER2			
			8.3.3.6-Confirmatory issue involving evaluation of penetrations' ability to withstand failure of overcurrent protection device	Resolved SSER7			
			8.3.3.1.1-Confirmatory issue involving submergence of electrical equipment as result of a LOCA	Resolved SSER13			
			8.3.3.2.2-Confirmatory issue to revise FSAR to reflect requirements of shared safety systems	Resolved SSER13			

1982 SER	10 CFR / GDC Note 2	CURRENT SRP TITLE Note 1	ISSUES from 1982 SER	Later SSERS	Guidance (GL, Bulletins , other)	Regulatory Guides Notes 1, 2	Additional Information
			8.3.1.6 LC – 12 Diesel generator reliability qualification testing at normal operating temperature	Resolved SSER2			
8.3.2.1 to 8.3.2.4, 8.3.3.1 to 8.3.3.6	*50.55a, 50.63, 50.65(a)(4)* GDC 2, 4, 5, 17, 18, 33, 34, 38, 41, 44, 50 *NUREG-0737 I.D.3*	8.3.2 - DC Power Systems (Onsite)	8.3.2.4-Confirmatory issue to include diesel generator design analysis in FSAR	Resolved SSER2		1.6, 1.32, 1,47, 1.53, 1.63, 1.75, 1.81, 1.106, 1.118, 1.128, 1.129, *1.153, 1.155, 1.160, 1.182,*	
			8.3.3.2.2-Confirmatory issue to revise FSAR to reflect requirements of shared safety systems	Resolved SSER13			
			8.3.2.2 – LC – Dc monitoring and annunciation system	Resolved SSER13			
			8.3.3.2.4 LC – Possible sharing of dc control power to ac switchgear	Resolved SSER3			
			8.3.3.3 LC – Testing of associated circuits	Resolved SSER3			
			8.3.3.3 LC – Testing of non-class 1E cables	Resolved SSER3			
			8.3.3.4 LC – Low	Resolved			

1982 SER	10 CFR / GDC Note 2	CURRENT SRP TITLE Note 1	ISSUES from 1982 SER	Later SSERS	Guidance (GL, Bulletins , other)	Regulatory Guides Notes 1, 2	Additional Information
			temperature overpressure protection power supplies, II.G.1	SSER7			
			8.3.3.6 LC – Testing of reactor coolant pump breakers	Resolved SSER2			
NA	*50.63, 50.65, GDC 17, 18*	8.4 Station Blackout	Not addressed in SER		GL 06-02	*1.9, 1.155, 1.160, 1.182*	SE for both units - March 18, 1993

Chapter 9 – Auxiliary Systems

9.1.1	*10 CFR 50.68 GDC 62*	9.1.1 – Criticality Safety of Fresh and Spent Fuel Storage and Handling	No open issues			*1.13R2*	
9.1.2	10 CFR 20.1101(b), 50.68 GDC 2, 4, 5, 61, 63	9.1.2 – New and Spent Fuel Storage	No open issues		Bulletin 84-03, 94-01	1.13, 1.29, 1.115, 1.117	
9.1.3	10 CFR 20.1101(b), GDC 2, 4, 5, 61, 63	9.1.3 - Spent Fuel Pool Cooling and Cleanup System	No open issues		SQN	1.13, 1.26, 1.29, 1.52, 8.8	
9.1.4	GDC 2, 5, 61, 62	9.1.4 - Light Load Handling System (Related to Refueling)	No open issues			1.29	
9.1.4	GDC *1*, 2, 4, 5	9.1.5 - Overhead Heavy Load Handling Systems	LC – Control of heavy loads (NUREG-0612)	Resolved SSER13	SQN	1.13, 1.29	
9.1.4	10 CFR 100.11 GDC 61	15.7.5 -Spent Fuel Cask Drop Accidents	No open issues			1.25	
9.2.1	GDC 2, 4, 5, 44, 45, 46	9.2.1 - Station Service Water System	No open issues in SER. SSER18 concludes	Resolved for Unit 1 in	GL89-13, 89-13 Supp 1, 91-13,	1.29	

1982 SER	10 CFR / GDC Note 2	CURRENT SRP TITLE Note 1	ISSUES from 1982 SER	Later SSERS	Guidance (GL, Bulletins , other)	Regulatory Guides Notes 1, 2	Additional Information
			ERCW does not conform to GDC 5 for two-unit operation.	SSER18 (Key Assumptions letter)	96-06, 96-06 Supp 1		
9.2.2	GDC 2, 4, 5, 44, 45, 46	9.2.2 - Reactor Auxiliary Cooling Water Systems	9.2.2-Confirmatory issue to relocate component cooling booster pumps above PMF level before receipt of an OL	Resolved SSER5 for Unit 1 (IR 390/84-20)	GL96-06, 96-06 Supp 1 SQN	1.29, *1.153, 1.155*	
9.2.3	GDC 2, 5	9.2.3 - Demineralized Water Makeup System	No open issues		SQN	1.29	
9.2.4	GDC 60	9.2.4 - Potable and Sanitary Water Systems	No open issues		SQN		
9.2.5	GDC 2, 5, 44, 45, 46	9.2.5 - Ultimate Heat Sink	No open issues		SQN	1.27, 1.72	
9.2.6	*10 CFR 50.63* GDC 2, 5, 44, 45, 46, 60	9.2.6 - Condensate Storage Facilities	No open issues		SQN	1.29, *1.143, 1.155, 1.59, 1.102, 1.76, 1.117*	
9.3.1	*10 CFR 50.63* GDC 1, 2, 5	9.3.1 - Compressed Air System	No open issues		GL88-14 SQN	1.155	
9.3.2	10 CFR 20.1101(b) GDC 1, 2, 13, 14, 26, 41, 60, 63, 64 NUREG 0737 III.D.1.1	9.3.2 - Process and Post-Accident Sampling Systems	9.3.2 LC – Post-Accident Sampling System	Resolved SSER14		1.21, 8.8, 1.26, 1.29, 1.97	
9.3.3	GDC 2, 4, 60	9.3.3 - Equipment and Floor Drainage System	No open issues				
9.3.4	*10 CFR 50.63(a)(2)* NUREG 0737 III.D.1.1	9.3.4 - Chemical and Volume Control System (PWR) (Including Boron	No open issues		Bulletin 80-18, 88-04 GL80-21, 89-04	1.26, 1.29, *1.155*	

1982 SER	10 CFR / GDC Note 2	CURRENT SRP TITLE Note 1	ISSUES from 1982 SER	Later SSERS	Guidance (GL, Bulletins , other)	Regulatory Guides Notes 1, 2	Additional Information
	GDC 1, 2, 5, 14, 29, 33, 35, 60, 61	Recovery System)					
9.4.1	*10 CFR 50.63* GDC 2, 4, 5, 19, 60	9.4.1 - Control Room Area Ventilation System	No open issues		SQN	1.29, 1.78*R1*, 1.52*R3*, *1.155*	
9.4.2	GDC 2, 5, 60, 61	9.4.2 - Spent Fuel Pool Area Ventilation System	No open issues		SQN	1.29, 1.52, 1.140, 1.13, *1.25*	
9.4.3	GDC 2, 5, 60	9.4.3 - Auxiliary and Radwaste Area Ventilation System	No open issues		SQN	1.29, 1.140, *1.52*	
9.4.4	GDC 2, 5, 60	9.4.4 - Turbine Area Ventilation System	No open issues		SQN	1.29, 1.140, *1.52*	
9.4.5	*10 CFR 50.63* GDC 2, 4, 5, 17, 60	9.4.5 - Engineered Safety Feature Ventilation System	No open issues		SQN	1.29, 1.140, 1.52, *1.155*	
9.5.1.1 to 9.5.1.9	10 CFR 50.48, Appendix R GDC 3, 5, *19, 23*	9.5.1 - Fire Protection Program	9.5.1.2-Outstanding issue for Fire Protection Program	Resolved SSERs 18, 19	SQN	*1.189R1, 1.174R1*	
			9.5.1.3 – Confirmatory issue – Electrical penetrations documentation				
			9.5.1.8 LC – Fire protection program				
9.5.2.1, 9.5.2.2	*Part 50 Appendix E, 50.47(a)(8), 50.55a, 73.45(e)(2)(iii), (g)(4)(i) 73.46(f), 73.55(e)*	9.5.2 - Communications Systems	9.5.2 LC – Performance testing of communications system	Resolved SSER5		*1.189R1, 1.180*	

1982 SER	10 CFR / GDC Note 2	CURRENT SRP TITLE Note 1	ISSUES from 1982 SER	Later SSERS	Guidance (GL, Bulletins , other)	Regulatory Guides Notes 1, 2	Additional Information
	and (f)						
9.5.3	None	9.5.3 - Lighting Systems	No open issues				
9.5.4.1, 9.5.4.2	GDC 2, 4, 5, 17	9.5.4 - Emergency Diesel Engine Fuel Oil Storage and Transfer System	9.5.4.1-Outstanding issue for staff to complete review to determine if diesel generator auxiliary support systems can perform their design safety functions under all conditions, after receipt of all requested information	Resolved SSER5	NUREG/Cr-0660	1.137	
			9.5.4.1-Confirmatory issue to include required language in operating instruction to ensure no-load and low-load operation is minimized and revise operating procedures to address increased diesel generator load after it has run for an extended period of time at low or no load	Resolved SSER5			
			9.5.4.1-Outstanding issue on definition of engine mounted piping	Resolved SSER5			

1982 SER	10 CFR / GDC Note 2	CURRENT SRP TITLE Note 1	ISSUES from 1982 SER	Later SSERS	Guidance (GL, Bulletins , other)	Regulatory Guides Notes 1, 2	Additional Information
			9.5.4.2-Outstanding issue to design skid-mounted piping and components from the day tank to the diesel engine as seismic Category I and to ASME Section III, Class 3	Resolved SSER5			
			9.5.4.2-Confirmatory issue to provide missile protection for fuel oil storage tank vent lines	Resolved SSER5			
			9.5.4.1 LC – Diesel Generator reliability	Resolved SSER5			
9.5.5	GDC 2, 4, 5, 17, 44, 45, 46	9.5.5 - Emergency Diesel Engine Cooling Water System	9.5.5-Outstanding issue to design engine cooling water system piping and components for all engines up to the engine interface, including auxiliary skid mounted piping, to ASME Section III, Class 3	Resolved SSER5	NUREG/Cr-0660	1.115, 1.117	
9.5.6	GDC 2, 4, 5, 17	9.5.6 - Emergency Diesel Engine Starting System	9.5.6-Outstanding issue to design engine air-starting system piping components for all engines up to the engine interface, including auxiliary skid mounted piping, to ASME Section III, Class 3	Resolved SSER5		1.115, 1.117	

1982 SER	10 CFR / GDC Note 2	CURRENT SRP TITLE Note 1	ISSUES from 1982 SER	Later SSERS	Guidance (GL, Bulletins , other)	Regulatory Guides Notes 1, 2	Additional Information
9.5.7	GDC 2, 4, 5, 17	9.5.7 - Emergency Diesel Engine Lubrication System	9.5.7-Outstanding issue to perform additional mod, or provide justification for acceptability of proposed mod, to ensure lubrication of all wearing parts of the diesel engine either on an interim or continuous basis	Resolved SSER5		1.115, 1.117	
			9.5.7-Outstanding issue to design standby diesel engine lube oil system piping and components up to the engine interface, including skid mounted piping, to ASME Section III, Class 3	Resolved SSER5			
			9.5.7-Outstanding issue to provide a more detailed description of the lubricating oil system and a description of the diesel engine crankcase explosion protection features	Resolved SSER5			
9.5.8	GDC 2, 4, 5, 17	9.5.8 - Emergency Diesel Engine Combustion Air Intake and Exhaust System	9.5.8-Outstanding issue to design standby diesel engine combustion air intake and exhaust system piping and components up	Resolved SSER5		1.115,1.117	

1982 SER	10 CFR / GDC Note 2	CURRENT SRP TITLE Note 1	ISSUES from 1982 SER	Later SSERS	Guidance (GL, Bulletins , other)	Regulatory Guides Notes 1, 2	Additional Information
			to the engine interface to ASME Section III, Class 3 and recommendations of RG 1.26				

Chapter 10 – Steam and Power Conversion System

10.2.1, 10.2.2	GDC 4	10.2 - Turbine Generator	No open issues			1.68	
10.2.2	GDC 4	10.2.3 - Turbine Rotor Integrity	No open issues				
10.3.1 to 10.3.4	*50.63*, GDC 2, 4, 5, 34	10.3 - Main Steam Supply System	10.3.4 LC – Secondary water chemistry monitoring and control program	Resolved for Unit 1 only SSER5 placed in TS administration section – same resolution for WBN2	GL 86-09 SQN	1.29, 1.115, 1.117, *1.155*	
10.3.3	Part 50, Appendix B, 50.55a, GDC 1, 35	10.3.6 - Steam and Feedwater System Materials	No open issues		GL 89-08 SQN	1.37, *1.50, 1.84,* 1.71	
10 4.1	*Part 50 Appendix I*, GDC 60	10.4.1 - Main Condensers	No open issues				
10.4.2	*Part 50 Appendix I*, GDC 60	10.4.2 - Main Condenser Evacuation System	No open issues				
10.4.3	GDC 60	10.4.3 - Turbine Gland Sealing System	No open issues				
10.4.4	GDC 4, 34	10.4.4 - Turbine Bypass	No open issues				

1982 SER	10 CFR / GDC Note 2	CURRENT SRP TITLE Note 1	ISSUES from 1982 SER	Later SSERS	Guidance (GL, Bulletins , other)	Regulatory Guides Notes 1, 2	Additional Information
		System					
10.4.5	GDC 4	10.4.5 - Circulation Water System	No open issues		SQN		
10.4.6	GDC 14	10.4.6 - Condensate Cleanup System	No open issues				
10.4.7	GDC 2, 4, 5, 44, 45, 46	10.4.7 - Condensate and Feedwater System	No open issues		GL 89-08	1.29	
10.4.8	GDC 1, 2, 13, 14	10.4.8 - Steam Generator Blowdown System (PWR)	No open issues			1.26, 1.29, 1.143	
10.4.9	*50.62, 50.63* GDC 2, 4, 5, 19, 34, 44, 45, 46 NUREG-0737 II.E.1.1,	10.4.9 - Auxiliary Feedwater System (PWR)	No open issues		B 85-01, GL 88-14	1.29, *1.117, 1.102, 1.59, 1.62, 1.76, 1.155*	

Chapter 11 - Radioactive Waste Management

11.1	10 CFR 20, 50 Appendix I GDC 60	11.1 - Source Terms	No open issues			1,140, 1.110, 1.112	
11.2	10 CFR 20.1302,20.1406, 50.34a, Appendix I Sections II.A and II.D GDC 60, 61	11.2 - Liquid Waste Management System	No open issues		SQN	1.143, 1.110, *1.70, 1.11, 1.33, 1.113, 1.112, 1.109*	
11.3	10 CFR 20.1302,20.1406,	11.3 - Gaseous Waste Management System	No open issues		SQN	1.140, 1.143, *1.11, 1.33,*	

1982 SER	10 CFR / GDC Note 2	CURRENT SRP TITLE Note 1	ISSUES from 1982 SER	Later SSERS	Guidance (GL, Bulletins , other)	Regulatory Guides Notes 1, 2	Additional Information
	50.34a, Appendix I Sections II.B, II.C and II.D GDC 3, 60, 61					*1.52, 1.70, 1.109, 1.110, 1.111, 1.112*	
11.4	10 CFR 20.1302,20.1301(e) , 20.1406, 50.34a, Appendix I Sections II.A IIB, IIC, and II.D GDC 60, *61*, 63 *10 CFR 61.55 and 61.56*	11.4 - Solid Waste Management System	No open issues		GL 80-09, 81-38, 81-39	1.143, *8.8, 8.10*	
11.5	10 CFR 20.1302,20.1301(e) , , 50.34a, 50.36(a), Appendix I , GDC 60, 63, 64	11.5 - Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems	No open issues		GL 80-09, 81-38, 81-39 B 80-10, 79-19	1.143, *1.33, 1.70, 1.11, 1.110, 1.112*	
11.7		NA NUREG -0737 items	LC – (6a) Accident monitoring instrumentation II.F.1 – Noble Gas monitor	Resolved SSER5			
			LC – (6b) Accident monitoring instrumentation II.F.1 – Iodine particulate sampling	Resolved SSER6			
			LC – Primary coolant outside containment III.D.1.1	Resolved for Unit 1 only SSER10 TS issue for waste gas disposal			

1982 SER	10 CFR / GDC Note 2	CURRENT SRP TITLE Note 1	ISSUES from 1982 SER	Later SSERS	Guidance (GL, Bulletins , other)	Regulatory Guides Notes 1, 2	Additional Information
				system – same resolution for WBN2			

Chapter 12 – Radiation Protection

12.2	10 CFR 19.12, 20.1101	12.1 - Assuring that Occupational Radiation Exposures are As Low As Reasonably Achievable	No open issues			1.8, 1.33, 1.70, 8.8, 8.10, *8.27*	
12.3	10 CFR 20.1201, .1202, .1203, .1204, .1206, .1207, .1301, .1801 GDC *19*, 61	12.2 - Radiation Sources	No open issues			1.4, 1.7, 1.112, *1.183*	
12.4	10 CFR 1101(b), .1201, .1202, .1203, .1004, .1701, .1301, .1302, .1406 GDC 19, 61, *63* *50.68*	12.3, 12.4 - Radiation Protection Design Features	No open issues		SQN	1.4, 1.7, 1.52, 1.69, 1.70, 1.97, *1.183*, 8.8, 8.10, 8.19, *8.25*, *8.38*	
12.5, 12.6	10 CFR 20, 19.12, *GDC 64* 71.5	12.5 - Operational Radiation Protection Program	12.6-Outstanding issue involving Health Physics Program	Resolved SSER10		1.8, 1.33, 1.97, 8.4, 8.6, 8.7, 8.8, 8.10, 8.13, 8.15, 8.20, *8.25*, 8.26, 8.27,	

1982 SER	10 CFR / GDC Note 2	CURRENT SRP TITLE Note 1	ISSUES from 1982 SER	Later SSERS	Guidance (GL, Bulletins , other)	Regulatory Guides Notes 1, 2	Additional Information
						8.28, 8.29, *8.32, 8.34, 8.35, 8.36, 8.38*	
12.7		NA NUREG 0737 items	LC – (6c) Accident monitoring instrumentation – containment radiation monitor	Resolved for Unit 1 only SSER5 (IR 390/84-23)			

Chapter 13 – Conduct of Operations

13.1.1	10 CFR 50.40(b)	13.1.1 - Management and Technical Support Organization	No open issues			1.8, 1.68	
13.1.2, 13.1.3	10 CFR 50.40(b), 50.54 i thru m	13.1.2, 13.1.3 - Operating Organization	13.1.3 LC – Use of experienced personnel during startup	Resolved SSER8		1.8, 1.33, 1.114	
13.2.1	10 CFR 50.54 i thru m, 55.4, 55.31, 55.41	13.2.1 - Reactor Operator Requalification Program, Reactor Operator Training	No open issues		SQN	1.8, *1.149*	
13.2.2	10 CFR 19.12, *26.21, 26.22,* 50.34(a) and (b), 50.40(b), 50.120, *Appendix E II.F, IV.F*	13.2.2 - Non-Licensed Plant Staff Training	No open issues		GL 86-04	1.8, 1.149	
13.3	*10 CFR 50.33*, .34, 50.47, *100.1,*	13.3 - Emergency Planning	13.3 LC – Emergency Preparedness III.A.1,	Resolved SSER13	Multiple GLs, Bulletins and	*1.23, 1.97,* 1.70, *1.101,*	

1982 SER	10 CFR / GDC Note 2	CURRENT SRP TITLE Note 1	ISSUES from 1982 SER	Later SSERS	Guidance (GL, Bulletins , other)	Regulatory Guides Notes 1, 2	Additional Information
	100.3, 50.72(a)(3), (a)(4), (c)(3), 73.71(a), Appendix E		III.A.2, III.A.2		RISs	*4.7, 5.62*	
13.4		13.4 - Operational Programs	13.4 LC – Independent Safety Engineering Group I.B.1.2	Resolved for Unit 1 only SSER8 TS issue – same resolution for WBN2			
13.5.1 13.5.2	10 CFR 50.40(b), 50.54(l), 50.34(a)(6) and (10), 50.34(b)(6)(iv) and (v)	13.5.1 - Administrative Procedures	13.5.2-Outstanding issue involving operating, maintenance and emergency procedures	Resolved SSER9	SQN	1.33	
			13.5.2 LC – Review of power ascension test procedures and emergency operating procedures by the NSSS vendor I.C.7	Resolved SSER10	GL 82-02, 82-12, 83-14, 89-23, 90-03, 91-16		
			13.5.2 LC – Modifications to Emergency Operating instructions	Resolved SSER10			
13.5.3		NA NUREG 0737 items	13.5.3 LC – Report on outage of emergency core cooling system II.K.3.17	Resolved SSER3			
13.6	10 CFR 26, 73.2, 73.55, 73.56, 73.57, Part 37	13.6 - Physical Security	13.6-Outstanding issue to file appropriate revision to the physical security plan	Resolved SSER15		5.12, 5.20, 5.44, 5.54	

1982 SER	10 CFR / GDC Note 2	CURRENT SRP TITLE Note 1	ISSUES from 1982 SER	Later SSERS	Guidance (GL, Bulletins , other)	Regulatory Guides Notes 1, 2	Additional Information
	Appendix B and C		13.6.4 LC – Physical security of fuel in containment	Resolved SSER10			
			13.6 Physical Security Plan				

Chapter 14 – Initial Test Program

1982 SER	10 CFR / GDC Note 2	CURRENT SRP TITLE Note 1	ISSUES from 1982 SER	Later SSERS	Guidance (GL, Bulletins , other)	Regulatory Guides Notes 1, 2	Additional Information
SER 14	10 CFR 50.34(b)(6)(iii), 30.53(c), Appendix J Section III.A.4	14.2 - Initial Plant Test Program – Design Certification and new License Application	14.0-Confirmatory issues- Availability of test procedures 60 days before test.	Resolved SSER3		*1.16*, 1.68, 1.68.2, 1.68.3, 1.20, 1.30, 1.37, 1.52, 1.56, 1.72, *1.78, 1.116*, 1.128, 1.139, 1.140	
			14.0-Confirmatory issue- FSAR references to Regulatory Guides.	Resolved SSER3			
			14.0-Confirmatory issue- Additional systems to be tested as part of the initial test program	Resolved SSER3			

1982 SER	10 CFR / GDC Note 2	CURRENT SRP TITLE Note 1	ISSUES from 1982 SER	Later SSERS	Guidance (GL, Bulletins , other)	Regulatory Guides Notes 1, 2	Additional Information
			Unit 2 issue to verify capability of each common station service transformer to carry load required to supply ESF loads of 1 unit under LOCA condition in addition to power required for shutdown on non-accident unit				
			14.2 LC - Initial Test Program	Resolved SSER7			
Section deleted		14.3 - Standard Plant Designs, Initial Test Program – Final Design Approval (FDA)					

Chapter 15 – Accident Analysis

15.1	Part 20, Part 50, 50.46, Part 100 GDC 2, 4, 5, 10, 13, 15, 17, 19, 20, 25, 26, 27, 28, 29, 31, 34, 35, 55, 60, 61	15 – Introduction – Transient and Accident Analysis	No open issues				
15.2	GDC 10, *13*, 15, *17*, *20*, 26	15.3.1-15.3.2 - Loss of Forced Reactor Coolant Flow Including Trip of Pump Motor and Flow	No open issues			*1.105, 1.53*	

1982 SER	10 CFR / GDC Note 2	CURRENT SRP TITLE Note 1	ISSUES from 1982 SER	Later SSERS	Guidance (GL, Bulletins , other)	Regulatory Guides Notes 1, 2	Additional Information
		Controller Malfunctions					
15.2.1	GDC 10, *13*, 15, *17*, 26	15.2.1 - 15.2.5 - Loss of External Load; Turbine Trip; Loss of Condenser Vacuum; Closure of Main Steam Isolation Valve (BWR); and Steam Pressure Regulator Failure (Closed)	No open issues			1.105, *1.53*	
15.2.1	GDC 10, *13*, 15, 26	15.2.6 - Loss of Nonemergency AC Power to Station Auxiliaries	No open issues			1.105, 1.53	
15.2.1	GDC 10, *13*, 15, *17*, 26 NUREG 0737 II.K.2.19	15.2.7 - Loss Normal Feedwater Flow	No open issues			1.105, 1.53, *1.206*	
15.2.2, 15.2.3	GDC 10, 13, 15, 20, 26	15.1.1- 15.1.4 - Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve	No open issues			1.105, 1.53	
15.2.3	GDC 10, *13*, 15, 26	15.5.1-15.5.2 - Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory	No open issues		RIS 2005-29	1.53, 1.105	

1982 SER	10 CFR / GDC Note 2	CURRENT SRP TITLE Note 1	ISSUES from 1982 SER	Later SSERS	Guidance (GL, Bulletins , other)	Regulatory Guides Notes 1, 2	Additional Information
15.2.4.1	GDC 10, *13, 17*, 20, 25	15.4.1 - Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition	No open issues				
15.2.4.2	GDC 10, *13, 17*, 20, 25	15.4.2 - Uncontrolled Control Rod Assembly Withdrawal at Power	No open issues				
15.2.4.3	GDC 10, *13*, 20, 25	15.4.3 - Control Rod Maloperation (System Malfunction or Operator Error)	No open issues				
NA	GDC 10, *13*, 15, 20, 26, 28	15.4.4-15.4.5 Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate	Not addressed in SER			1.105, 1.53, 1.70	
15.2.4.4	GDC 10, *13*, 15, 26	15.4.6 - Chemical and Volume Control Systems Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant (PWR)	15.2.4.4 Outstanding issue for evaluation of Boron dilution and single failure criteria	Resolved SSER4			
15.2.4.5	Part 100 GDC 13	15.4.7 - Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position	No open issues				

1982 SER	10 CFR / GDC Note 2	CURRENT SRP TITLE Note 1	ISSUES from 1982 SER	Later SSERS	Guidance (GL, Bulletins , other)	Regulatory Guides Notes 1, 2	Additional Information
15.2.6	*10 CFR 100.11, 50.67* GDC *13*, 28	15.4.8 - Spectrum of Rod Ejection Accidents (PWR)	No open issues			1.77, *1.183, 1.195*	
15.3.1	10 CFR 50 Appendix K, 50.46, 100, *50.67* GDC *13*, 35 NUREG 0737 II.K.3.5, II.K.3.25, II.K.3.30, II.K.3.31	15.6.5 - Loss-of-Coolant Accidents Resulting From Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary	No open issues		GL 85-12 SQN	*1.157*	
15.3.2	GDC *13, 17*, 27, 28, 31, 35	15.1.5 - Steam System Piping Failures Inside and Outside Containment (PWR)	No open issues				
15.3.3	10 CFR 100 GDC *13, 17*, 27, 28, 31, 35 NUREG 0737 II.E.1.1, II.E.1.2	15.2.8 - Feedwater System Pipe Breaks Inside and Outside Containment (PWR)	No open issues				
15.3.4, 15.3.5	10 CFR 100 GDC *17*, 27, 28, 31	15.3.3-15.3.4 - Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break	No open issues				
15.3.6	*10 CFR 50.62, 50.46 GDC 12, 14, 16, 35, 38, 50*	15.8 - Anticipated Transients Without Scram	LC - Anticipated Transients Without Scram (Generic Letter 83-28, Item 4.3)	Resolved SSER5			
15.4.1	10 CFR 100.11	15.6.5.A - Radiological Consequences of a Design Basis Loss-of-	No open issues			1.4	

1982 SER	10 CFR / GDC Note 2	CURRENT SRP TITLE Note 1	ISSUES from 1982 SER	Later SSERS	Guidance (GL, Bulletins , other)	Regulatory Guides Notes 1, 2	Additional Information
		Coolant Accident Including Containment Leakage Contribution					
15.4.2, 15.4.6	10 CFR 100.11	15.6.5.B - Radiological Consequences of a Design Basis Loss-of-Coolant Accident: Leakage from Engineered Safety Feature Components Outside Containment	No open issues			1.4, 1.7, 1.52	
15.4.2	10 CFR 100.11	15.1.5.A - Radiological Consequence of Main Steam Line Failures Outside Containment of a PWR	No open issues			1.4	
15.4.3	10 CFR 100.11	15.6.3 - Radiological Consequences of Steam Generator Tube Failure	LC – Steam Generator tube rupture	Resolved SSER12 and 14		1.4	
15.4.4	10 CFR 100	15.4.8.A - Radiological Consequences of a Control Rod Ejection Accident (PWR)	No open issues			1.77	
15.4.5	10 CFR 100.11 GDC 61	15.7.4 - Radiological Consequences of Fuel Handling Accidents	No open issues			1.25	
15.4.6	10 CFR 100 GDC 55	15.6.2 - Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant	No open issues			1.11	

1982 SER	10 CFR / GDC Note 2	CURRENT SRP TITLE Note 1	ISSUES from 1982 SER	Later SSERS	Guidance (GL, Bulletins , other)	Regulatory Guides Notes 1, 2	Additional Information
		Outside Containment					
15.4.7	10 CFR 20 GDC 60	15.7.3 - Postulated Radioactive Releases Due to Liquid-Containing Tank Failures	No open issues				
15.5.1- 15.5.2		NA NUREG 0737 items	LC – Effect of high pressure injection for small beak LOCA with no auxiliary feedwater II.K.2.13	Resolved SSER4			
			LC – Voiding in the reactor coolant system II.K.2.17	Resolved SSER4			
15.5.3	GDC 10, *13*, 15, 26 NUREG 0737 II.K.3.25	15.6.1 - Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve or a BWR Pressure Relief Valve	LC – PORV isolation system II.K.3.1, II.K.3.2	Resolved SSER5	GL 85-12, 86-05, 86-06	1.105, 1.53	
15.5.4 – 15.4.5		NA NUREG 0737 items	LC – Automatic trip of reactor coolant pumps during a small break LOCA	Resolved SSER4			
			LC – Revised small break LOCA analysis	Resolved SSER5			

Chapter 16 – Technical Specifications

16	10 CFR 50.34(b)(6)(vi), 50.36	16 – Technical Specifications	No open issues				

1982 SER	10 CFR / GDC Note 2	CURRENT SRP TITLE Note 1	ISSUES from 1982 SER	Later SSERS	Guidance (GL, Bulletins , other)	Regulatory Guides Notes 1, 2	Additional Information
NA	*10 CFR 50.36*	16.1 - Risk Informed Decision Making: Technical Specifications	Not addressed in SER			*1.177, 1.174*	

Chapter 17 – Quality Assurance

1982 SER	10 CFR / GDC Note 2	CURRENT SRP TITLE Note 1	ISSUES from 1982 SER	Later SSERS	Guidance (GL, Bulletins , other)	Regulatory Guides Notes 1, 2	Additional Information
17.1, 17.2	Part 50 Appendix B, 50.34(a.7), 50.55a, 50.55e, GDC 1	17.1 – Quality Assurance (QA) During the Design and Construction Phase	No open issues			1.8, 1.26, 1.28, 1.29, 1.30, 1.37, 1.38, 1.39, 1.58, 1.64, 1.74, 1.88, 1.94, 1.116, 1.123, 1.144, 1.146	
17.3, 17.4	Part 50 Appendix B, 50.34(b.6ii), GDC 1	17.2 – QA During the Operations Phase	Outstanding issue QA program	Resolved SSER2, Updated SSER5, Resolved SSER13		1.33	
17.3	Part 50 Appendix B	17.3 – Quality Assurance Program Description	No open issues		GL 89-02	1.8, 1.26, 1.28, 1.29, 1.33, 1.152, 1.143, 1.36, 1.54, 4.15	
NA	10 CFR 50.65	17.6 – Maintenance Rule	Not addressed in SER			1.160, 1.182	

1982 SER	10 CFR / GDC Note 2	CURRENT SRP TITLE Note 1	ISSUES from 1982 SER	Later SSERS	Guidance (GL, Bulletins , other)	Regulatory Guides Notes 1, 2	Additional Information

Chapter 18 – Control Room Design Review*

18	50.34(f)(1)(i), (f)(2), (f)(3)(i), (f)(3)(vii), (i) to (m), 50.120 NUREG 0737 I.D.1	18 - Human Factors Engineering	LC – Detailed Control Room Design review I.D.1	Resolved for Unit 1in SSER15 with onsite audit of Unit 1 control room improvements – same resolution for WBN2		1.9, 1.22, 1.47, 1.62, 1.81, 1.97, 1.105, 1.174, 1.177, 1.187	
			LC – Make Safety Parameter Display System operable prior to startup from the first refueling outage	Open item for Unit 2- resolution requires a functional system before fuel load and on-line testing after Unit 2 is operational then an operational certification (GL89-06)			

***SRP Chapter 18 was retitled to Human Factors Engineering in 1984**

1982 SER	10 CFR / GDC Note 2	CURRENT SRP TITLE Note 1	ISSUES from 1982 SER	Later SSERS	Guidance (GL, Bulletins , other)	Regulatory Guides Notes 1, 2	Additional Information

Chapter 19 – Severe Accidents

NA	**NA for Part 50**	19.0 - Probabilistic Risk Assessment and Severe Accident Evaluation for New Reactors	Not Applicable				
	Part 50	19.1- Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities	Not addressed in SER			*1.174, 1.200R1*	
	50.11, 50.12, 50.90	19.2 – Review of Risk Information Used to Support Permanent Plant Specific Changes to the Licensing Basis: General Guidance	Not addressed in SER			*1.174R1, 1.200R1*	

Item	Title	Applicable FSAR Section	Status	Comments

Chapter 20 – Generic Communications

IEB 73-01	Faulty Overcurrent Trip Delay Device in Circuit Breakers for Engineered Safety Systems		Closed	
IEB 73-02	Malfunction of Containment Purge Supply Valve Switch	6.2.4	Closed	
IEB 73-03	Defective Hydraulic Snubbers and Restraints	3.9.3	Closed	OL @ time
IEB 73-04	Defective Bergen-Patterson Hydraulic Shock Absorbers	3.9.3	Closed	
IEB 73-05	Manufacturing Defect in BWR Control Rods		NA	BWR
IEB 73-06	Inadvertent Criticality in a BWR		NA	BWR
IEB 74-01	Valve Deficiencies	3.9.3, 3.9.6	Closed	
IEB 74-02	Truck Strike Possibility		NA	Dated issue
IEB 74-03	Failure of Structural or Seismic Support Bolts on Class I Components	3.9.3	Closed	
IEB 74-04	Malfunction of Target Rock Safety Relief Valves		Closed	BWR
IEB 74-05	Shipment of an Improperly Shielded Source		NA	Not generation
IEB 74-06	Defective Westinghouse (W) Type W-2 Control Switch Component		Closed	
IEB 74-07	Personnel Exposure – Irradiation Facility		NA	Not generation
IEB 74-08	Deficiency in the ITE Molded Case Circuit Breakers, Type HE-3		Closed	
IEB 74-09	Deficiencies in GE Model 4KV Magne-Blast Circuit Breakers		Closed	
IEB 74-10	Failures in 4 inch Bypass Pipe at Dresden		NA	BWR
IEB 74-11	Improper Wiring of Safety Injection Logic at Zion 1 & 2		Closed	
IEB 74-12	Incorrect Coils in W Type SG Relays at Trojan		Closed	
IEB 74-13	Improper Factory Wiring on GE Motor Control Centers at Fort Calhoun		Closed	
IEB 74-14	BWR Relief Valve Discharge to Suppression Pool		NA	BWR
IEB 74-15	Misapplication of Cutler-Hammer Three Position Maintained Switch		Closed	

Item	Title	Applicable FSAR Section	Status	Comments
	Model No. 10250T			
IEB 74-16	Improper Machining of Pistons in Colt Industries (Fairbanks-Morse) Diesel Generators		Closed	
IEB 75-01	Through-Wall Cracks in Core Spray Piping at Dresden-2		NA	BWR
IEB 75-02	Defective Radionics Radiograph Exposure Devices and Source Changes		NA	Not generation
IEB 75-03	Incorrect Lower Disc Spring and Clearance Dimension in Series 8300 and 8302 ASCO Solenoid Valves	3.9.6	Closed	
IEB 75-04	Cable Fire at BFNPP	9.5.1	Closed	
IEB 75-05	Operability of Category I Hydraulic Shock and Sway Suppressors	3.9.3	Closed	
IEB 75-06	Defective W Type OT-2 Control Switches		Closed	
IEB 75-07	Exothermic Reaction in Radwaste Shipment		NA	Not generation
IEB 75-08	PWR Pressure Instrumentation		Closed	
IEB 76-01	BWR Isolation Condenser Tube Failure		Closed/ NA	By inspection - BWR
IEB 76-02	Relay Coil Failures		Open	
IEB 76-03	Relay Malfunctions – GE Type STD Relays		Closed	
IEB 76-04	Cracks in Cold Water Piping at BWRs		NA	BWR
IEB 76-05	Relay Failures		Closed	
IEB 76-06	Diaphragm Failures in Air Operated Auxiliary Actuators for Safety/Relief Valves	3.9.3, 3.9.6	Closed	
IEB 76-07	Crane Hoist Control Circuit Modifications		Closed	
IEB 76-08	Teletherapy Units		NA	Not generation
IEB 77-01	Pneumatic Time Delay Relay Setpoint Drift		Closed	
IEB 77-02	Potential Failure Mechanism in Certain W AR Relays with Latch Attachments		Closed	
IEB 77-03	On-Line Testing of the W Solid State Protection System		Open	
IEB 77-04	Calculation Error Affecting Performance of a System for Controlling	6.2	Closed	

Item	Title	Applicable FSAR Section	Status	Comments
	pH of Containment Sump Water Following a LOCA			
IEB 77-05 & 77-05 A	Electrical Connector Assemblies		Closed	
IEB 77-06	Potential Problems with Containment Electrical Penetration Assemblies	3.10, 3.11	Closed	
IEB 77-07	Containment Electrical Penetration Assemblies at Nuclear Power Plants under Construction	3.10, 3.11	Closed	
IEB 77-08	Assurance of Safety and Safeguards During an Emergency – Locking Systems		Closed	
IEB 78-01	Flammable Contact – Arm Retainers in GE CR120A Relays		Closed	
IEB 78-02	Terminal Block Qualification		Closed	
IEB 78-03	Potential Gas Mixture Accumulations Associated with BWR Offgas System Operations		NA	BWR
IEB 78-04	Environmental Qualification of Certain Stem Mounted Limit Switches Inside Reactor Containment	3.11	Closed	
IEB 78-05	Malfunctioning of Circuit Breaker Auxiliary Contact Mechanism		Closed	
IEB 78-06	Defective Cutler-Hammer Type M Relays and DC Coils		Closed	
IEB 78-07	Protection Afforded by Air-Line Respirators and Supplied-Air Hoods		NA	Site
IEB 78-08	Radiation Levels from Fuel Element Transfer Tubes		NA	Site
IEB 78-09	BWR Drywell Leakage Paths Associated with Inadequate Drywell Closures		NA	BWR
IEB 78-10	Bergen-Patterson Hydraulic Shock Suppressor Accumulator Spring Coils	3.9.3	Closed	
IEB 78-11	Examination of Mark I Containment Torus Welds		NA	BWR
IEB 78-12	Atypical Weld Material in Reactor Pressure Vessel Welds	5.4	Closed	
IEB 78-13	Failures in Source Heads Kay Ray Gauge Models 7050, 7050B, 7051, 7051B, 7060, 7060B, 7061 and 7061B		NA	Not generation
IEB 78-14	Deterioration of Buna-N Components in ASCO Solenoids		NA	BWR
IEB 79-01	Environmental Qualification of Class 1E Equipment	3.11	Closed	
IEB 79-02	Pipe Support Base Plate Designs Using Concrete Expansion Anchor	3.9.3	Open	

Item	Title	Applicable FSAR Section	Status	Comments
	Bolts			
IEB 79-03	Longitudinal Weld Defects in ASME SA-312 Type 304 SS Pipe Spools Fabricated by Youngstown Welding & Engineering	3.9.3, 6.1.1	Closed	
IEB 79-04	Incorrect Weights for Swing Check Valves Manufactured by Velan Engineering	3.7, 3.9	Closed	
IEB 79-05	Nuclear Incident at TMI		NA	B & W
IEB 79-06	Review of Operational Errors and System Misalignments Identified During the Three Mile Island Incident		Closed	
IEB 79-07	Seismic Stress Analysis of Safety-Related Piping	3.9.3	Closed	
IEB 79-08	Events Relevant to BWRs Identified During TMI Incident		NA	BWR
IEB 79-09	Failure of GE Type AK-2 Circuit Breaker in Safety Related Systems		Closed	
IEB 79-10	Requalification Training Program Statistics		NA	Site
IEB 79-11	Faulty Overcurrent Trip Device in Circuit Breakers for Engineering Safety Systems		Closed	
IEB 79-12	Short Period Scrams at BWR Facilities		NA	BWR
IEB 79-13	Cracking in Feedwater Piping	3.9.3, 6.1.1	Closed	OL @ time
IEB 79-14	Seismic Analysis for As-Built Safety-Related Piping Systems	3.9.3	Open	
IEB 79-15	Deep Draft Pump Deficiencies	3.9.3, 3.9.6	Closed	
IEB 79-16	Vital Area Access Controls		NA	Site
IEB 79-17	Pipe Cracks in Stagnant Borated Water Systems at PWR Plants	3.9	Closed	
IEB 79-18	Audibility Problems Encountered on Evacuation of Personnel from High-Noise Areas		NA	Site
IEB 79-19	Packaging of Low-Level Radioactive Waste for Transport and Burial		NA	Site
IEB 79-20	Packaging, Transport and Burial of Low-Level Radioactive Waste		NA	Site
IEB 79-21	Temperature Effects on Level Measurements		Open	
IEB 79-22	Possible Leakage of Tubes of Tritium Gas Used in Time Pieces for Luminosity		NA	Site
IEB 79-23	Potential Failure of Emergency Diesel Generator Field Exciter Transformer		Closed	
IEB 79-24	Frozen Lines	6.3	Open	

Item	Title	Applicable FSAR Section	Status	Comments
IEB 79-25	Failures of W BFD Relays in Safety Related Systems		Closed	
IEB 79-26	Boron Loss from BWR Control Blades		NA	BWR
IEB 79-27	Loss of Non-Class 1E I & C Power System Bus During Operation		Open	
IEB 79-28	Possible Malfunction of NAMCO Model EA180 Limit Switches at Elevated Temperatures		Closed	
IEB 80-01	Operability of ADS Valve Pneumatic Supply		NA	BWR
IEB 80-02	Inadequate QA for Nuclear Supplied Equipment		NA	BWR
IEB 80-03	Loss of Charcoal from Standard Type II, 2 Inch Tray Adsorber Cells	6.5	Closed	
IEB 80-04	Analysis of a PWR Main Steam Line Rupture with Continued Feedwater Addition	3.6, 5.5.9, 6.2, 10.3, 10.4	Open	
IEB 80-05	Vacuum Condition Resulting in Damage to Chemical Volume Control System Holdup Tanks	9.3.4	Open	
IEB 80-06	Engineered Safety Features Reset Control	6.2.2, 6.3, 6.5	Open	
IEB 80-07	BWR Jet Pump Assembly Failure		NA	BWR
IEB 80-08	Examination of Containment Liner Penetration Welds	3.8	Closed	
IEB 80-09	Hydramotor Actuator Deficiencies		Closed	
IEB 80-10	Contamination of Non-radioactive System and Resulting Potential for Unmonitored, Uncontrolled Release of Radioactivity to Environment		Open	
IEB 80-11	Masonry Wall Design	3.8.4	Open	
IEB 80-12	Decay Heat Removal System Operability	10.4.7, 10.4.8	Closed	
IEB 80-13	Cracking in Core Spray Spargers		NA	BWR
IEB 80-14	Degradation of Scram Discharge Volume Capability		NA	BWR
IEB 80-15	Possible Loss of Emergency Notification System with Loss of Offsite Power		NA	Site
IEB 80-16	Potential Misapplication of Rosemount Models 1151 and 1152 Pressure Transmitters With Either "A" or "D" Output Codes		Closed	
IEB 80-17	Failure of 76 of 185 Control Rods to Fully Insert During a Scram at a		NA	BWR

Item	Title	Applicable FSAR Section	Status	Comments
	BWR			
IEB 80-18	Maintenance of Adequate Minimum Flow Thru Centrifugal Charging Pumps Following a Secondary Side High Energy Rupture	6.3	Open	
IEB 80-19	Mercury-Wetted Matrix Relay in Reactor Protective Systems		Closed	
IEB 80-20	Failure of W Type W-2 Spring Return to Neutral Control Switches		Open	
IEB 80-21	Valve Yokes Supplied by Malcolm Foundry Co.	3.9.3, 3.9.6	Closed	
IEB 80-22	Automation Industries, Model 200-520-008 Sealed-Source Connectors		NA	Not generation
IEB 80-23	Failures of Solenoid Valves Manufactured by Valcor Eng. Corp	3.9.3, 3.9.6	Closed	
IEB 80-24	Prevention of Damage Due to Water Leakage Inside Containment		Open	
IEB 80-25	Operating Problems with Target Rock SR Valves at BWRs		NA	BWR
IEB 81-01	Surveillance of Mechanical Snubbers	3.9.3	Closed	
IEB 81-02	Failure of Gate Type Valves to Close Against Differential Pressure	3.9.6	Closed	
IEB 81-03	Flow Blockage to Cooling Water by Asiatic Clams and Mussels	9.2.1	Closed	
IEB 82-01	Surveillance of Mechanical Snubbers	3.9.3	Closed	
IEB 82-02	Degradation of Threaded Fasteners in the Reactor Coolant Pressure Boundary of PWR Plants	3.9.3	Closed	
IEB 82-03	Stress Corrosion Cracking in Thick-Wall, Large Diameter, Stainless Steel, Recirculation System Piping at BWR Plants		NA	BWR
IEB 82-04	Deficiencies in Primary Containment Electrical Penetration Assemblies	3.10, 3.11	Closed	
IEB 83-01	Failure of Trip Breakers (Westinghouse DB-50) to Open on Automatic Trip Signal		Closed	
IEB 83-02	Stress Corrosion Cracking in Large-Diameter Stainless Steel Recirculation System Piping at BWR Plants		NA	BWR
IEB 83-03	Check Valve Failures in Raw Water Cooling Systems of Diesel Generators		NA	Addressed by IST for CP holders
IEB 83-04	Failure of the Undervoltage Trip Function of Reactor Trip Breakers		Closed	
IEB 83-05	ASME Nuclear Code Pumps and Spare Parts Manufactured by	3.9.3, 3.9.6,	Closed	

Item	Title	Applicable FSAR Section	Status	Comments
	Hayward Tyler Pump Co	17.1		
IEB 83-06	Nonconforming Material Supplied by Tube-Line	3.9.3, 5.2.1, 6.1.1, 17.1	Closed	
IEB 83-07	Apparently Fraudulent Products Sold by Ray Miller, Inc	17.1	Closed	
IEB 83-08	Electrical Circuit Breakers With Undervoltage Trip in Safety Related Applications other than the Reactor Trip System		Closed	
IEB 84-01	Cracks in BWR Mark 1 Containment Vent Headers		NA	BWR
IEB 84-02	Failure of GE Type HFA Relays In Use In Class 1E Safety Systems		Closed	
IEB 84-03	Refueling Cavity Water Seal	9.1.1, 9.1.2	Open	
IEB 85-01	Steam Binding of Auxiliary Feedwater Pumps	10.4.9	Closed	
IEB 85-02	Undervoltage Trip Attachment of W DB-50 Type Reactor Trip Breakers		Open	
IEB 85-03	Motor Operated Valve Common Mode Failures During Plant Transients due to Improper Switch Settings	3.9.3, 3.9.6	Closed	
IEB 86-01	Minimum Flow Logic Problems That Could Disable RHR Pumps		NA	BWR
IEB 86-02	Static "O" Ring Differential Pressure Switches		Closed	
IEB 86-03	Potential Failure of Multiple ECCS Pumps Due to Single Failure of Air-Operated Valve in Minimum Flow Recirculation Line	6.3	Closed	
IEB 86-04	Defective Teletherapy Timer That May Not Terminate Treatment Dose		NA	Not generation
IEB 87-01	Thinning of Pipe Walls in Nuclear Power Plants	3.9.3, 6.1	Closed	
IEB 87-02	Fastener Testing to Determine Conformance with Applicable Material Specifications	17.1	Closed	
IEB 88-01	Defects in W Circuit Breakers		Closed	
IEB 88-02	Rapidly Propagating Fatigue Cracks in Steam Generator Tubes	5.5.2	Open	
IEB 88-03	Inadequate Latch Engagement in HFA Type Latching Relays Manufactured by General Electric (GE) Company		Closed	
IEB 88-04	Potential Safety-Related Pump Loss	5.5.7, 6.3	Open	
IEB 88-05	Nonconforming Materials Supplied by Piping Supplies, Inc. and West Jersey Manufacturing Company	3.9.3, 6.1, 17.1	Open	

Item	Title	Applicable FSAR Section	Status	Comments
IEB 88-06	Actions to be Taken for the Transfer of Model No. SPEC 2-T Radiographic Exposure Device		NA	Not generation
IEB 88-07	Power Oscillations in BWRs		NA	BWR
IEB 88-08	Thermal Stresses in Piping Connected to Reactor Cooling Systems	3.9.3, 5.5.7, 6.3	Open	
IEB 88-09	Thimble Tube Thinning in Westinghouse Reactors	5.2	Open	
IEB 88-10	Nonconforming Molded-Case Circuit Breakers		Open	
IEB 88-11	Pressurizer Surge Line Thermal Stratification	3.9.3	Open	
IEB 89-01	Failure of Westinghouse Steam Generator Tube Mechanical Plugs	5.5.2	Open	
IEB 89-02	Stress Corrosion Cracking of High-Hardness Type 410 Stainless Steel Preloaded Bolting in Anchor Darling Model S350W Swing Check Valves or Valves of Similar Nature	3.9.3, 3.9.6, 6.1	Open	
IEB 89-03	Potential Loss of Required Shutdown Margin During Refueling Operations	9.1	Closed	
IEB 90-01	Loss of Fill-Oil in Transmitters Manufactured by Rosemount		Open	
IEB 94-01	Potential Fuel Pool Draindown Caused by Inadequate Maintenance Practices at Dresden		NA	Shutdown plants
IEB 96-01	Control Rod Insertion Problems (PWR)	4.2	Open	
IEB 96-02	Movement of Heavy Loads over Spent Fuel, over Fuel in the Reactor, or over Safety-Related Equipment	3.8.6, 9.1	Open	
IEB 96-03	Potential Plugging of ECCS Suction Strainers by Debris in BWRs		NA	BWR
IEB 96-04	Chemical, Galvanic or Other Reactions in Spent Fuel Storage and Transportation Casks		NA	Not generation
IEB 97-01	Potential for Erroneous Calibration, Dose Rate, or Radiation Exposure Measurements with Certain Victoreen Model 530 and 531SI Electrometers, Dose Meters		NA	Not generation
IEB 97-02	Puncture Testing of Shipping Packages Under 10 CFR Part 71		NA	Not generation
IEB 01-01	Circumferential Cracking of Reactor Pressure Vessel (RPV) Head Penetration Nozzles	5.2.3, 5.2.4, 5.2.5, 5.4	Open	

Item	Title	Applicable FSAR Section	Status	Comments
IEB 02-01	RPV Head Degradation and Reactor Coolant Pressure Boundary Integrity	5.2.3, 5.2.4, 5.2.5, 5.4	Open	
IEB 02-02	RPV Head and Vessel Head Penetration Nozzle Inspection Program	5.4	Open	
IEB 03-01	Potential Impact of Debris Blockage on Emergency Sump Recirculation	6.2	Open	
IEB 03-02	Leakage from RPV Lower Head Penetrations and Reactor Coolant Pressure Boundary Integrity	5.2.3 – 5.2.5, 5.4	Open	
IEB 03-03	Potentially Deficient 1-inch Valves for Uranium Hexaflouride Cylinders		NA	Not generation
IEB 03-04	Rebaselining of Data in the Nuclear Management and Safeguards System		NA	Not generation
IEB 04-01	Inspection of Alloy 82/182/600 Materials Used in the Fabrication of Pressurizer Penetrations and Steam Space Piping Connections at PWRs	3.9.3, 6.1	Open	
IEB 05-01	Material Control and Accounting of Reactors and Spent Fuel Storage Facilities		NA	Site
IEB 05-02	Emergency Preparedness and Response Actions for Security Based Events		NA	Site
GL 77-06	Questionnaire Related to Steam Generators		NA	OL @ time
GL 77-07	Reliability of Standby Diesel Generator Units		NA	OL @ time
GL 78-02	Asymmetric Loads Background and Revised Request for Additional Information	3.1, 3.9.3, 3.9.5	Open	
GL 78-03	Request for Information on Cavity Annulus Seal Ring	9.1.1, 9.1.2	Closed	See IEB 84-03
GL 78-04	Steam Generator Operating History		NA	OL @ time
GL 78-09	Multiple Subsequent Actuations of Safety Relief Valves Following an Isolated Event		NA	BWR
GL 78-15	Request for Information on Control of Heavy Loads Near Spent Fuel		NA	Addressed by later BU/GLs

Item	Title	Applicable FSAR Section	Status	Comments
GL 78-34	Reactor Vessel Atypical Weld Material	5.4.2, 6.1	Open	
GL 79-20	Cracking in Feedwater Lines	3.9.3, 6.1	Open	
GL 79-24	Multiple Equipment Failures in Safety Related Systems		NA	OL @ time
GL 79-36	Adequacy of Station Electric Distribution Systems Voltages	8.2, 8.3.1	Open	
GL 79-42	Potentially Unreviewed Safety Question on Interaction Between Non-Safety Grade Systems and Safety Grade Systems		NA	OL @ time
GL 79-46	Containment Purge and Venting During Normal Operation		NA	OL @ time
GL 79-52	Radioactive Release at No. Anna Unit 1 and Lessons Learned		NA	OL @ time
GL 79-57	Acceptance Criteria for Mark I Long Term Program		NA	BWR
GL 79-58	ECCS Calculations on Fuel Cladding		NA	OL @ time
GL 80-02	QA Requirements Regarding Diesel Generator Fuel Oil		NA	Site
GL 80-13	Qualification of Safety Related Electrical Equipment		NA	OL @ time
GL 80-14	LWR Primary Coolant System Pressure Isolation Valves	3.9.3, 3.9.6, 5.5	Open	
GL 80-18	Crystal River 3 Reactor Trip From Approximately 100% Full Power		NA	B & W
GL 80-21	Vacuum Condition Resulting in Damage to Chemical Volume Control System Holdup Tanks	9.3.4	Open	
GL 80-32	Information Request on Category I Masonry Walls Employed by Plants Under CP and OL Review		NA	Addressed by IEB 80-11
GL 80-34	Clarification of NRC Requirements for Emergency Response Facilities at Each Site		NA	Site
GL 80-38	Summary of Certain Non-Power Reactor Physical Protection Requirements		NA	Not generation
GL 80-46/47	A-12, Fracture Toughness and Additional Guidance on Potential for Low Fracture Toughness and Laminar Tearing on PWR Steam Generator Coolant Pump Supports	3.1, 3.9.3	Open	
GL 80-53	Decay Heat Removal Capability		NA	OL @ time
GL 80-60	Request for Information Regarding Evaluation Times		NA	Site
GL 80-77	Refueling Water Level – Technical Specifications Changes		NA	Site
GL 80-84	BWR Scram System		NA	BWR

Item	Title	Applicable FSAR Section	Status	Comments
GL 80-90	NUREG-0737, TMI (Prior and future GLs, with the exception of certain discrete scopes, have been screened into NUREG list for those applicable to Watts Bar 2)		Open	See NUREG list
GL 80-95	A-10, NUREG-0619, "BWR Feedwater Nozzle and Control Rod Drive Return Line Cracking"		NA	BWR
GL 80-113	Control of Heavy Loads		Closed	See later GLs, IEBs
GL 81-03	NUREG-0313, Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping		NA	BWR
GL 81-04	Emergency Procedures and Training for Station Blackout		Closed	Subsumed by SB
GL 81-07	Control of Heavy Loads	9.1.4	Open	
GL 81-12	Fire Protection Rule (and prior related GLs)		NA	OLs prior to 1/1/79
GL 81-14	Seismic Qualification of Auxiliary Feedwater Systems		NA	OL @ time
GL 81-20, -30, -34, -35	Safety Concerns Associated With Pipe Breaks in the BWR Scram System		NA	BWR
GL 81-21	Natural Circulation Cooldown	5.5	Open	
GL 82-28	Inadequate Core Cooling Instrumentation System	4.4	Open	
GL 82-33	Supplement to NUREG-0737, Requirements for Emergency Response Capability	7.5.2	Open	
GL 83-08	Modification of Vacuum Breakers on Mark I Containments		NA	BWR
GL 83-10c	Automatic Trip of Reactor Coolant Pumps	5.5	Closed	
GL 83-28	Required Actions Based on Generic Implications of Salem ATWS Events (Prior and future GLs, with the exception of certain discrete scopes, have been screened into the following list): 1.2 – Post Trip Review Data and Information Capability 2.1 – Equipment Classification and Vendor Interface (Reactor Trip	15.2	 Closed Closed	

Item	Title	Applicable FSAR Section	Status	Comments
	System Components) 2.2 – Equipment Classification and Vendor Interface (All SR Components) 3.1 – Post-Maintenance Testing (Reactor Trip System Components) 3.2 – Post-Maintenance Testing (All SR Components) 4.1 – Reactor Trip System Reliability (Vendor Related Modifications) 4.3 – Reactor Trip System Reliability (Automatic Actuation of Shunt Trip Attachment)		Open Closed Closed Open Closed	
GL 84-09	Recombiner Capability Requirements of 10 CFR 50.44(c)(3)(ii)		NA	BWR
GL 84-11	Inspection of BWR Stainless Steel Piping		NA	BWR
GL 84-23	Reactor Vessel Water Level Instrumentation in BWRs		NA	BWR
GL 84-24	Certification of Compliance to 10 CFR 50.49	3.11	Closed	Incorporated into license review
GL 85-02	Recommended Actions Stemming From NRC Integrated Program for the Resolution of Unresolved Safety Issues Regarding Steam Generator Tube Integrity	5.5.2	Open	
GL 85-11	Completion of Phase II of "Control of Heavy Loads at Nuclear Power Plants"		Closed	See GL 81-07
GL 85-12	Implementation of TMI Item II.K.3.5	15.5.4	Open	
GL 86-05	Implementation of TMI Item II.K.3.5		NA	B & W
GL 86-06	Implementation of TMI Item II.K.3.5		NA	CE
GL 87-02 & 03	Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors, USI A-46		NA	Previously operated plants
GL 87-05	Request for Additional Information on Assessment of License Measures to Mitigate and/or Identify Potential Degradation of Mark I Drywells		NA	BWR

Item	Title	Applicable FSAR Section	Status	Comments
GL 87-06	Periodic Verification of Leak Tight Integrity of Pressure Isolation Valves		NA	OL @ time
GL 87-12	Loss of Residual Heat Removal While The Reactor Coolant System is Partially Filled	5.4.7	Closed	
GL 88-01	NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping		NA	BWR
GL 88-03	Resolution of GSI 93, Steam Binding of Auxiliary Feedwater Pumps	10.4.9	Closed	
GL 88-05	Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR plants	5.2	Open	
GL 88-07	Modified Enforcement Policy Relating to 10 CFR 50.49, EQ of Electrical Equipment Important to Safety for Nuclear Power Plants	3.11	Closed	To Special Program
GL 88-11	NRC Position on Radiation Embrittlement of Reactor Vessel Material and its Impact on Plant Operations	5.2.3, 5.2.4, 5.2.5, 5.4	Open	
GL 88-14	Instrument Air Supply System Problems Affecting Safety-Related Equipment	9.3.1	Open	
GL 88-17	Loss of Decay Heat Removal	4.4, 6.2,6.3	Open	
GL 88-20	Individual Plant Examination for Severe Accident Vulnerability		Open	
GL 89-04	Guidelines on Developing Acceptable Inservice Testing Programs	3.9.3, 3.9.6	Open	
GL 89-06	TMI Action Plan Item I.D.2 – Safety Parameter Display System	7.5.2	Open	
GL 89-07	Power Reactor Safeguards Contingency Planning for Surface Vehicle Bombs		NA	Site
GL 89-08	Erosion/Corrosion-Induced Pipe Wall Thinning	6.6, 10.3, 10.4.7	Open	
GL 89-10	Safety-Related Motor-Operated Valve Testing and Surveillance	6.9	Open	
GL 89-13	Service Water System Problems Affecting Safety Related Equipment	9.2.1	Open	
GL 89-15	Emergency Response Data System		NA	Site
GL 89-16	Installation of a Hardened Wetwell Vent		NA	BWR
GL 89-19	Request for Actions Related to Resolution of Unresolved Safety Issue A-47 "Safety Implication of Control Systems in LWR Nuclear Power Plants" Pursuant to 10 CFR 50.54(f)	7.7, 10.3	Open	
GL 89-20	Protected Area Long Term Housekeeping		NA	Site

Item	Title	Applicable FSAR Section	Status	Comments
GL 89-22	Potential For Increased Roof Load Due to Changes in Maximum Precipitation	2.3	Open	
GL 90-06	Resolution of Generic Issues 70, "PORV and Block Valve Reliability," and 94, "Additional LTOP Protection for PWRs"	5.2.2, 5.5.7	Open	
GL 91-06	Resolution of Generic Issue A-30, "Adequacy of Safety Related DC Power Supplies"		NA	OL @ time
GL 91-11	Resolution of Generic Issues A-48, "LCOs for Class 1E Vital Instrument Buses", and 49, "Interlocks and LCOs for 1E Tie Breakers"		NA	OL @ time
GL 91-13	Request for Information Related to Resolution of Generic Issue 130, "Essential Service Water System Failures @ Multi-Unit Sites"		NA	Selected plants
GL 92-01	Reactor Vessel Structural Integrity	5.2.3, 5.2.4, 5.2.5, 5.4	Open	
GL 92-04	Resolution of Issues Related to Reactor Vessel Water Level Instrumentation in BWRs		NA	BWR
GL 92-08	Thermo-Lag 330-1 Fire Barriers	9.5.1	Open	
GL 93-01	Emergency Response Data System Test Program		NA	Site
GL 93-04	Rod Control System Failure and Withdrawal of Rod Control Cluster Assemblies, 10 CFR 50.54(f)	4.4	Open	
GL 94-02	Long Term Solutions and Upgrade of Interim Operating Recommendations for Thermal/Hydraulic Instabilities in BWRs		NA	BWR
GL 94-03	IGSCC of Core Shrouds in BWRs		NA	BWR
GL 95-01	NRC Staff Technical Position on Fire Protection for Fuel Cycle Facilities		NA	Not generation
GL 95-03	Circumferential cracking of Steam Generator (SG) Tubes	5.5.2	Open	
GL 95-05	Voltage Based Repair Criteria for W SG Tubes Affected by Outside Diameter Stress Corrosion Cracking	5.5.2	Open	
GL 95-07	Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves	6.9	Open	
GL 96-01	Testing of Safety-Related Circuits		Open	

Item	Title	Applicable FSAR Section	Status	Comments
GL 96-03	Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits	5.2.2, 5.2.4	Open	
GL 96-05	Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves	6.9	Open	
GL 96-06	Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions	9.2.1, 9.2.2	Open	
GL 97-01	Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations	5.2.3, 5.2.4, 5.2.5	Open	
GL 97-04	Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps	6.2.2, 6.3	Open	
GL 97-05	SG Tube Inspection Techniques	5.5.2	Open	
GL 97-06	Degradation of SG Internals	5.5.2	Open	
GL 98-01	Year 2000 Readiness of Computer Systems at Nuclear Power Plants		NA	Site
GL 98-02	Loss of Reactor Coolant Inventory and Associated Potential for Loss of Emergency Mitigation Functions While in a Shutdown Condition	5.5.7, 6.3	Open	
GL 98-04	Potential for Degradation of the ECCS and the Containment Spray System After a LOCA Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment	6.1.2, 6.2.2, 6.3	Open	
GL 99-02	Laboratory Testing of Nuclear Grade Activated Charcoal		NA	Site
GL 03-01	Control Room Habitability	6.4	Closed	
GL 04-01	Requirements for SG Tube Inspection	5.5.2	Open	
GL 04-02	Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at PWRs	6.2.2	Open	
GL 06-01	SG Tube Integrity and Associated Technical Specifications	5.5.2	Open	
GL 06-02	Grid Reliability and the Impact on Plant Risk and the Operability of Offsite Power	8.0, 8.1, 8.2, 8.3	Open	
GL 06-03	Potentially Nonconforming Hemyc and MT Fire Barrier Configurations	9.5.1	Open	
GL 07-01	Inaccessible or Underground Power Cable Failures that Disable Accident Mitigation Systems or Cause Plant Transients	8.3.1	Open	

Item	Title	Applicable FSAR Section	Status	Comments
NUREG 0612	Control of Heavy Loads	9.1.4	Open	Due to GL 81-07 commitment
NUREG 0737	TMI Items:			
	I.C.1 Short term accident and procedure review	13.5	Open	
	I.C.7 NSSS vendor revision of procedures	13.5	Open	
	I.C.8 Pilot monitoring of selected emergency procedures for NTOLs	13.5	Open	
	I.D.1 CRDR	6.4	Open	
	I.D.2 Plant-safety-parameter-display console	7.5.2	Open	
	II.B.1 Reactor-coolant-vent-system	5.5.6	Open	
	II.B.2 Plant shielding	3.11	Open	
	II.B.3 Post-accident sampling	9.3.2	Open	
	II.D.1 Relief and safety valve test requirements	5.2.2	Open	
	II.D.3 Valve position indication	5.2.2, 6.3, 7.5	Open	
	II.E.1.1 Auxiliary feedwater system evaluation, modifications	10.4.9	Open	
	II.E.1.2 Auxiliary feedwater system initiation and flow	7.3, 7.5	Closed	
	II.E.3.1 Emergency power for pressurizer heaters	8.3.1	Closed	
	II.E.4.2 Containment isolation dependability	6.2.4, 7.3	Closed	
	II.F.1.2. Accident-monitoring instrumentation			
	A. -Noble gas	7.5	Open	
	B. - Iodine/particulate sampling	7.5	Open	
	C. - Containment high range monitoring	7.5	Open	
	D. - Containment pressure	7.5	Open	
	E. - Containment water level		Open	

Item	Title	Applicable FSAR Section	Status	Comments
	F. - Containment Hydrogen	7.5	Open	
	II.F.2 Instrumentation for detection of inadequate core-cooling	7.5	Open	
	II.G.1 Power supplies for pressurizer relief valves, block valves and level indicators	4.4, 6.3, 7.5 7.4, 7.5,	Open	
	II.K.1.5 Review ESF valves	8.3.1	Closed	
	II.K.1.10 Operability status		Open	
	II.K.1.17 Trip per low-level B/S		Closed	
	II.K.3.1 Auto PORV isolation		Open	
	II.K.3.3 Reporting SV/RV failures/challenges	15.5.3	Open	
	II.K.3.5 Auto trip of RCPs	5.2.2	Open	
	II.K.3.9 PID controller	15.5.4	Open	
	II.K.3.10 Anticipatory trip at high power		Closed	
	II.K.3.17 ECCS outages		Open	
	II.K.3.25 Power on pump seals	6.3	Open	
	II.K.3.30 SB LOCA methods	9.2.2	Closed	
	II.K.3.31 Plant specific analysis	15.5.5	Open	
	III.D.1.1 Primary coolant outside containment	15.5.5	Open	
		5.5.7, 6.3,		
	III.D.3.3 In-plant I2 radiation monitoring	9.2	Open	
	III.D.3.4 Control-room habitability		Closed	
		6.4		

Chapter 21 - Corrective Action Programs and Special Programs

Description		Status for Unit 2
1.	CAP - Cable Issues	Open
2.	CAP - Cable Tray Supports	Open
3.	CAP - Conduit Supports	Open
4.	CAP - Design Baseline Verification Program	Open
5.	CAP - Electrical Issues	Open
6.	CAP - Equipment Seismic Qualification	Open
7.	CAP – Fire Protection	Open
8.	CAP - Hanger and Analysis Update Program	Open
9.	CAP - Heat Code Traceability	Open
10.	CAP - HVAC Duct Supports	Open
11.	CAP - Instrument Sensing Lines	Open
12.	CAP - Pre-start Test Program	Withdrawn. TVA will perform a Regulatory Guide 1.68 Preoperational Test Program for Unit 2
13.	CAP - QA Records	Open
14.	CAP - Q-List	Open
15.	CAP - Piece Parts	Open

Description		Status for Unit 2
16.	CAP - Seismic Analysis	Open
17.	CAP - Vendor Performance	Open
18.	CAP - Welding	Open
19.	SP - Concrete Quality	Complete
20.	SP - Containment Cooling	Open
21.	SP - Control Room Design Review	Open
22.	SP - Equipment Qualification	Open
23.	SP - Master Fuse List	Open
24.	SP - Mechanical EQ	Open
25.	SP - Micro. Induced Corrosion	Open
26.	SP - MELB Flooding	Open
27.	SP - Radiation Monitoring System	Open
28.	SP - Soil Liquefaction	Complete
29.	SP - Use-as-is CAQs	Open

Chapter 22 - Regulatory Guide Applicability for Watts Bar Unit 2

Regulatory Guide Title / Rev	Full or Partial Compliance	SRP Section
1.1 – Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps / Rev. 0	Full	6.3
1.4 – Assumptions used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors / Rev. 2	Partial	2.3.4, 6.2.1, 6.5.1, 6.5.2, 6.5.3, 12.2, 12.3, 12.4, 15.6.5, 15.6.3
1.6 – Independence Between Redundant Standby (onsite) Power Sources and Between Their Distribution Systems / Rev. 0	Full	8.3.1, 8.3.2
1.7 – Control of Combustible Gas Concentrations in Containment Following a Loss of Cooling Accident / Rev. 2	Partial	3.8.1, 3.8.2, 6.2.5, 6.5.1, 7.5, 12.2, 12.3, 12.4, 15.6.5
1.8 – Qualification and Training of Personnel for Nuclear Power Plants / Rev. 2	Partial	12.1, 12.5, 13.1.1, 13.1.2, 13.1.3, 13.2.1, 13.2.2, 17.1, 17.2
1.9 – Selection, Design and Qualification of Diesel Generator Used as Standby (onsite) Electric Power Systems at Nuclear Power Plants / Rev. 3	Partial	8.3.1, 8.4, 18
1.10 – Mechanical (Cadweld) Splices in Reinforcing Bars of Category I Concrete Structures / Rev. 1	Partial	
1.12 – Instrumentation for Earthquakes / Rev. 1	Full	3.7.4
1.13 – Spent Fuel Storage Facility Design Basis / Rev. 1	Partial	9.1.1, 9.1.2, 9.1.3, 9.1.5, 9.4.2
1.14 – Reactor Coolant Pump Flywheel Integrity / Rev. 1	Partial	5.4.1.1
1.15 – Testing of Reinforcing Bars for Category I Concrete Structures / Rev 1	Partial	
1.16 – Reporting of Operating Information, Appendix A Technical Specifications / Rev 4	Partial	14.2
1.20 – Comprehensive Vibration Assessment Program For Reactor Internals During Preoperational and Initial Startup Testing / Rev. 2	Partial	3.9.5, 14.2

Regulatory Guide Title / Rev	Full or Partial Compliance	SRP Section
1.21 – Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants / Rev. 1	Partial	9.3.2
1.22 – Periodic Testing of Protection System Actuation Functions / Rev.0	Full	7.2, 7.3, 7.9, 18
1.23 – Onsite Meteorological Programs / Rev. 0	Partial	2.3.2, 2.3.3, 2.3.4, 2.3.5, 13.3
1.24 – Assumptions Used for Evaluating Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure / Rev. 0	Full	2.3.4
1.25 – Assumptions Used for Evaluating Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors / Rev. 0	Partial	4.2, 6.5.1, 9.4.2, 15.7.4, 15.7.5
1.26 – Quality Group Classifications and Standards for Water, Steam, and Radioactive Waste Containing Components of Nuclear Power Plants / Rev. 3	Partial	3.2.2, 3.9.4, 5.2.1.1, 5.2.2, 5.4.11, 6.2.7, 9.1.3, 9.3.2, 9.3.4, 10.4.8, 17.1, 17.3
1.27 – Ultimate Heat Sink for Nuclear Power Plants / Rev.1	Partial	2.4.1- 2.4.9, 2.4.11, 2.4.12, 3.5.2, 9.2.5
1.28 – Quality Assurance Program Requirements – Design and Construction / Rev. 3	Full	2.5.4, 2.5.5, 17.1, 17.3
1.29 – Seismic Design Classification / Rev. 1	Partial	2.4.1- 2.4.11, 2.4.14, 3.2.1, 3.4.1, 3.9.4, 5.2.2, 5.2.5, 5.4.7, 5.4.11 6.2.4, 6.3, 9.1.2-9.1.5, 9.2.1- 9.2.3, 9.2.5, 9.3.2-9.3.4, 9.4.1-9.4.5, 10.3, 10.4.7-10.4.9, 17.1, 17.3
1.30 – Quality Assurance Requirements for the Installation, Inspection and Testing of Instrumentation and Electrical Equipment / Rev. 0	Partial	14.2, 17.1
1.31 – Control of Ferrite in Stainless Steel Weld Material / Rev. 3	Partial	4.5.2, 5.2.3, 5.3.1, 5.4.2.1, 6.1.1

Regulatory Guide Title / Rev	Full or Partial Compliance	SRP Section
1.32 – Criteria for Safety-related Electric Power Systems for Nuclear Power Plants / Rev. 2	Full	8.2, 8.3.1, 8.3.2
1.33 – Quality Assurance Program Requirements – Operation / Rev.2	Partial	11.2, 11.3, 11.5, 12.1, 12.5, 13.1.2, 13.1.3, 13.5.1, 172, 17.3
1.36 – Nonmetallic Thermal Insulation for Austenitic Stainless Steel / Rev 0	Partial	5.2.3, 5.4.2.1, 6.1.1, 17.3
1.37 – Quality Assurance Requirements for Cleaning Fluid Systems and Associated Equipment of Water Cooled Nuclear Power Plants / Rev. 0	Partial	3.13, 5.2.3, 5.3.1, 5.4.2.1, 6.1.1, 10.3.6, 14.2, 17.1
1.38 – Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage and Handling of Items for Water-Cooled Nuclear Power Plants / Rev. 2	Partial	17.1
1.39 – Housekeeping Requirements for Water-cooled Nuclear Power Plants / Rev. 2	Partial	17.1
1.40 – Qualification Tests of Continuous-Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants / Rev. 0	Full	
1.41 – Preoperational Testing of Redundant On-site Electric Power Systems to Verify Proper Load Group Assignments / Rev. 0	Full	
1.44 – Control of the use of Sensitized Stainless Steel / Rev.0	Partial	4.5.1, 4.5.2, 5.2.3, 5.3.1, 5.4.2.1, 6.1.1
1.45 – Reactor Coolant Pressure Boundary Leakage Detection Systems / Rev.0	Partial	3.6.3, 5.2.5
1.46 – Protection Against Pipe Whip Inside Containment / Rev.0	Partial	
1.47 – Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems / Rev.0	Full	6.3, 7.2, 7.3, 7.4, 7.5, 18
1.48 – Design Limits and Loading Combinations for Seismic Category I Fluid System Components / Rev.0	Partial	

Regulatory Guide Title / Rev	Full or Partial Compliance	SRP Section
1.50 – Control of Preheat Temperature for Welding of Low-Alloy Steel / Rev. 0	Partial	5.2.3, 5.3.1, 5.4.2.1, 6.1.1, 10.3.6
1.52 – Design, Testing and Maintenance Criteria for Post-Accident Engineered Safety Feature Atmospheric Cleanup System Air Filtration and Adsorption Units of Light Water-Cooled Nuclear Power Plants / Rev. 2	Partial	6.3, 6.4, 6.5.1, 6.5.3, 9.1.3, 9.4.1 – 9.4.5, 11.3, 12.3, 12.4, 14.2, 15.6.5B
1.53 – Application of the Single Failure Criterion to Nuclear Power Plants Protection Systems / Rev. 0	Full	8.3.1, 8.3.2, 9.2.2
1.54 – Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants / Rev. 0	Partial	17.3, 6.1.2
1.55 – Concrete Placement in Category I Structures / Rev. 0	Partial	
1.58 – Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel / Rev. 1	Partial	17.1
1.59 – Design Basis Floods for Nuclear Power Plants / Rev. 2	Partial	2.4.1- 2.4.10, 2.4.14, 9.2.6, 10.4.9
1.60 – Design Response Spectra for Seismic Design of Nuclear Power Plants / Rev. 1	Partial	2.5.2, 3.7.1, 3.7.2, 3.12, 4.2
1.61 – Damping Values for Seismic Design of Nuclear power Plants / Rev. 0	Partial	3.9.2
1.62 – Manual Initiation of Protective Actions / Rev.0	Full	10.4.9, 18
1.63 – Electrical Penetration Assemblies in Containment for Nuclear Power Plants / Rev. 3	Full	8.3.1, 8.3.2
1.64 – Quality Assurance Requirements for the Design of Nuclear Power Plants / Rev. 2	Partial	17.1
1.67 – Installation of Overpressure Protection Devices / Rev. 0	Partial	
1.68 – Initial Test Programs for Water-Cooled Nuclear Power Plants / Rev. 2	Partial	3.9.2, 4.4, 6.3, 10.2, 14.2, 13.1.1, 14.2
1.68.2 – Initial Startup Test Program to Demonstrate Remote Shutdown Capability for Water-Cooled Nuclear Power Plants / Rev. 1	Partial	14.2

Regulatory Guide Title / Rev	Full or Partial Compliance	SRP Section
1.68.3 – Preoperational Testing of Instrument and Control Air Systems / Rev. 0	Partial	14.2
1.70 – Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants – LWR Edition / Rev. 3	Partial	2.1.3, 2.2.3, 2.3.1, 2.3.4, 2.3.5, 2.4.1- 2.4.8, 3.7.1-3.7.4,3.8.1- 3.8.5, 7.1, 11.3, 11.5, 12.1, 12.3, 12.4, 13.3, 15.4.4
1.71 – Welder Qualification for Areas of limited Accessibility / Rev. 0	Partial	5.2.3, 5.4.2.1, 10.3.6
1.73 – Qualification Tests of Electric Valve Operators Installed Inside Containment of Nuclear Power Plants / Rev. 0	Full	
1.74 – Quality Assurance Terms and Definitions / Rev. 0	Partial	17.1
1.75 – Physical Independence of Electrical Systems / Rev. 2	Partial	7.2, 7.3, 7.9, 8.3.1, 8.3.2
1.76 – Design Basis for Nuclear Power Plants / Rev. 0	Partial	2.3.1, 3.3.2, 3.5.1.4, 3.5.3, 9.2.6, 10.4.9
1.77 – Assumptions used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors / Rev. 0	Partial	2.3.4, 4.2, 4.3, 15.4.8, 15.4.8A
1.78 – Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release / Rev. 0	Partial	2.2.1, 2.2.2, 2.3.4, 6.4, 9.4.1, 14.2
1.79 – Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors / Rev.1	Partial	6.3
1.81 – Shared Emergency and Shutdown Electrical Systems for Multi-Unit Nuclear Power Plants / Rev.1	Partial	3.8.1, 3.8.2, 18
1.82 – Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident / Rev. 0	Partial	5.4.7, 6.2.2, 6.3
1.83 - Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes / Rev. 1	Partial	
1.84 – Design and Fabrication Code Case Acceptability ASME Section III Division I / Rev 33	Partial	3.2.2, 3.8.1, 3.12, 3.13, 4.5.2, 5.2.1.2, 5.2.2, 5.2.3, 5.4.2.1, 6.1.1, 10.3.6
1.85 – Materials Code Case Acceptability ASME Section III Division I / - Withdrawn Incorporated into RG 1.84	NA	3.2.2, 4.5.1

Regulatory Guide Title / Rev	Full or Partial Compliance	SRP Section
1.88 – Collection, Storage, and Maintenance of Nuclear Power Plants Quality Assurance Records / Rev. 2	Partial	17.1
1.89 – Qualification of Class 1E Equipment for Nuclear power Plants / Rev. 1	Partial	3.10
1.91 – Evaluation of Explosions Postulated to Occur on Transportation Routes / Rev. 1	Partial	2.2.1-2.2.2, 3.5.1.5, 3.8.1, 3.8.4
1.92 – Combining Model Responses and Spatial Components in Seismic Response Analysis / Rev. 1	Partial	3.7.2, 3.12, 5.4.12
1.93 – Availability of Electric Power Sources / Rev. 0	Full	
1.94 – Quality Assurance Requirements for Installation, Inspection and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants / Rev. 1	Partial	17.1
1.95 – Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release / Rev. 1	Partial	2.2.1-2.2.2
1.97 – Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant and Evirons Conditions During and Following an Accident / Rev. 2	Partial	3.10, 6.2.1, 6.2.1.1B, 6.2.5, 7.5, 9.3.2, 12.3, 12.4, 12.5, 13.3, 18
1.99 – Radiation Embrittlement of Reactor Vessel Materials / Rev. 2	Partial	5.3.2- 5.3.3
1.100 – Seismic Qualification of Electrical Equipment for Nuclear Power Plants / Rev. 1	Partial	3.10, 5.4.12
1.101 – Emergency Planning and Preparedness for Nuclear Power Plants / Rev. 2	Partial	13.3
1.102 – Flood Protection for Nuclear Power Plants / Rev. 1	Partial	2.4.1-2.4.10, 2.4.14, 9.2.6, 10.4.9
1.105 – Instrument Setpoints for Safety-Related Systems / Rev. 1	Partial	2.4.1-2.4.10, 2.4.14, 9.3.6, 10.4.9
1.106 – Thermal Overload Protection for Electric Motors on Motor-Operated Valves / Rev. 1	Partial	8.3.1, 8.3.2

Regulatory Guide Title / Rev	Full or Partial Compliance	SRP Section
1.109 – Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluation Compliance with 10 CFR 50 Appendix I / Rev. 1	Partial	2.3.5, 11.2, 11.3
1.111 – Methods of Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light Water Cooled Reactors / Rev. 1	Partial	2.3.5, 11.3
1.112 – Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light Water Cooled Power Reactors / Rev. 0	Partial	2.3.5, 11.1, 11.2, 11.3, 11.5, 12.2
1.113 – Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purposes of Implementing Appendix I / Rev. 1	Partial	2.4.13, 11.2
1.114 – Guidance to Operators at the Controls and to Senior Operators in the Control Room of a Nuclear Power Unit / Rev. 2	Partial	13.1.2, 13.1.3
1.115 – Protection Against Low Trajectory Turbine Missiles / Rev. 1	Partial	3.5.1.1, 3.5.1.3, 3.5.2, 3.8.1, 3.8.4, 9.5.5- 9.5.8, 10.3
1.116 – Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems / Rev. 0	Partial	14.2, 17.1
1.117 – Tornado Design Classification / Rev. 1	Partial	3.5.1.3, 3.5.1.5, 3.5.2, 9.1.2, 9.3.1, 9.5.5- 9.5.8, 10.3, 10.4.9
1.118 – Periodic Testing of Electric Power and Protection Systems / Rev. 2	Partial	7.2, 7.3, 7.9, 8.3.1, 8.3.2
1.122 – Development of Floor Design Response Spectra for Seismic Design of Flood-Supported Equipment or Components / Rev. 1	Partial	3.7.2, 3.12
1.123 – Quality Assurance Requirements for Control Procurement of Items and Services for Nuclear Power Plants / Rev. 1	Partial	17.1

Regulatory Guide Title / Rev	Full or Partial Compliance	SRP Section
1.124 – Service Limits and Loading Combinations for Class 1 Linear Type Component Supports / Rev. 1	Full	3.9.3, 3.12
1.126 – An Acceptable Model and Related Statistical Methods for Analysis of Fuel Densification / Rev. 1	Partial	4.2, 4.3
1.127 – Inspection of Water Control Structures Associated with Nuclear Power Plants / Rev. 1	Partial	3.8.4, 3.8.5
1.128 – Installation Design and Installation of Large Lead Storage Batteries for Nuclear Power Plants / Rev. 0	Partial	3.8.2, 14.2
1.129 – Maintenance, Testing and Replacement of Large Lead Storage Batteries for Nuclear Power Plants / Rev. 1	Full	3.8.2
1.130 – Service Limits and Loading Combinations for Class 1 Plate-and-Shell Type Component Supports / Rev. 1	Partial	3.9.3, 3.12
1.133 – Loose Part Detection Program for the Primary System of Light Water Cooled Reactors / Rev. 1	Partial	4.4
1.137 – Fuel Oil Systems for Diesel Generators / Rev. 1	Partial	9.5.4
1.139 – Guidance for Residual Heat Removal / Rev. 0	Partial	14.2
1.140 – Design, Maintenance and Testing Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light Water Cooled Nuclear Power Plants / Rev. 1	Partial	6.5.1, 9.4.2, 9.4.3, 9.4.4, 9.4.5, 11.3, 14.2
1.141 – Containment Isolation Provisions for Fluid Systems / Rev. 0	Partial	
1.143 – Design Guidance for Radioactive Waste Management Systems, Structures and Components Installed in Light Water Cooled Nuclear Power Plants / Rev. 1	Partial	3.2.1, 3.8.3, 3.8.4, 9.2.6, 10.4.8, 11.2, 11.3, 11.4, 11.5, 17.3
1.144 – Auditing of Quality Assurance Programs for Nuclear Power Plants / Rev. 1	Partial	17.1
1.145 – Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants / Rev. 1	Partial	2.3.4

Regulatory Guide Title / Rev	Full or Partial Compliance	SRP Section
1.146 – Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants / Rev. 0	Partial	17.1
1.147 – Inservice Inspection Code Case Acceptability ASME Section XI Division 1 / Rev.	Partial	3.8.1, 3.8.2, 5.2.1.2, 5.2.4
1.150 – Ultrasonic Testing of Reactor Vessel Welds during Preservice and Inservice Examinations / Rev 1	Partial	5.2.4
1.152 – Criteria for Programmable Digital Computer Software in Safety-Related Systems of Nuclear Power Plants / Rev. 0	Partial	7.1, 7.2, 7.3, 7.4, 7.5, 7.6, 7.7, 7.8, 7.9, 17.3
1.153 – Criteria for Power, Instrumentation and Control Portions of Safety Systems / Rev. 0	Partial	8.3.1, 8.3.2, 9.2.2
1.155 – Station Blackout / Rev. 1	Partial	6.2.5, 6.3, 8.2, 8.3.1, 8.3.2, 8.4, 9.2.2, 9.2.6, 9.3.1,.9.3.4, 9.4.1, 9.4.5, 10.3, 10.4.9
1.158 – Qualification of Safety-Related Lead Storage Batteries for Nuclear Power Plants / Rev. 0	Partial	
1.189 – Fire Protection for Operating Nuclear Power Plants / Rev. 0	Partial	3.2.1, 7.4, 9.5.1, 9.5.2

Notes:

1. **The requirements that must be met before a plant can be licensed are defined in NRC regulations. Over the years, the NRC staff has prepared a number of guidance documents, such as regulatory guides and the Standard Review Plan that define methods that are acceptable to the NRC staff for meeting various requirements in the regulations. Except for a few regulatory guides that are specifically referenced in a regulation, these guidance documents are not requirements. The Standard Review Plan is not a substitute for the NRC's regulations and compliance is not required. Regulatory Guides are not substitutes for regulations and compliance with them is not required.**
2. **10 CFR sections, GDCs and Regulatory Guides introduced in the revised SRP are shown in italics.**
3. **Identification of relevant FSAR section in Chapter 20 is based on current FSAR entries, and future entries may impact other sections.**
4. **Technical Specification changes discussed in Generic Letters, and related to NUREG-0737, that may require license amendment are not included in this table.**

Attachment 2 - Outstanding Corrective Action Programs and Special Programs

This Attachment provides a description of the Watts Bar Nuclear Performance Plan (NPP) Corrective Action Programs (CAPs) and Special Programs (SPs), and a summary of their resolution for WBN Unit 1. TVA has evaluated these CAPs and SPs in order to determine their applicability to WBN Unit 2. These issues are listed in the order used in the NPP.

TVA will resolve the WBN Unit 2 CAPs and SPs consistent with NUREG-1232 (Volume 4), NUREG-0847 and applicable regulations. If, during this process, TVA determines that it is necessary to modify the criteria otherwise specified in NUREG-1232, it will submit such changes to the NRC for review and concurrence. TVA will use the procedures, databases and templates that were developed for WBN Unit 1 to the fullest extent practical for WBN Unit 2.

Corrective Action Programs:

1. Cable Issues

The Cable Issues CAP was initiated based on various employee concerns, conditions adverse to quality (CAQ) documents, and NRC findings related to cable installation and routing.

The Cable Issues CAP identified the following issues:

1. Silicone rubber insulated cables
2. Cable jamming
3. Cable support in vertical conduits
4. Cable support in vertical trays
5. Cable proximity to hot pipes
6. Cable pullbys
7. Cable bend radius
8. Cable splices
9. Cable sidewall bearing pressure
10. Pulling cables through 90-degree condulets and mid-route flexible conduits
11. Computerized Cable Routing System (CCRS) software and data base verification and validation

Each of these issues is discussed individually below:

<u>Silicone Rubber Insulated Cables</u>

The initial test program proposed by TVA for SQN in April of 1987 to address concerns with silicone rubber insulated cables, included hi-pot testing of silicone rubber insulated cables manufactured by American Insulated Wire (AIW), Rockbestos and Anaconda. The testing revealed a significant number of failures in AIW cables. TVA decided to replace AIW cables. Rockbestos and Anaconda cables were successfully tested at Wyle Laboratories for 40 year qualified life.

Cable Jamming

NRC identified the potential for undetected cable damage since TVA-WBN documents did not address cable jam ratio. When single conductors with unacceptable jam ratios are pulled into a conduit, the cable may align in a flat configuration with resultant jamming.

For Unit 1, Class 1E conduits were evaluated to identify those segments most likely to have experienced jamming during installation. These segments were ranked according to their calculated percent sidewall bearing pressure. A sampling of cables were removed and inspected, and no evidence of damage due to jamming was identified. The inspected cables included those with the highest calculated side wall bearing pressure and are considered to bound the lower ranked cables.

Cable Support in Vertical Conduits

NRC expressed a concern that cables in long vertical conduits were inadequately supported and that random failures due to cutting of the insulation and conductor creep may occur during normal service conditions, especially for silicone rubber insulated cables.

For Unit 1, TVA identified the critical cases of silicone rubber insulated cables in vertical conduits, with cable bearing pressure occurring at the edge of the condulet the determining factor. A comparison was made of WBN critical cases with those already tested at SQN. If SQN conduits enveloped WBN, no cable testing by WBN was performed. If SQN conduits did not envelope WBN, cable was replaced or in situ cable testing was performed. If the testing option was selected, any cable found unacceptable was replaced. Additionally, TVA evaluated Class 1E conduits containing cables of all insulation types using acceptance criteria that provided for cable supports to be added for the conduits in which the cable bearing pressure, conductor strength, or resultant loading imparted to the cable insulation, terminations or splices exceeded manufacturer's limits, as a result of the cable weight from the long vertical distance. In addition, cable installation specification and site procedures were revised to incorporate appropriate cable support requirements for cable installed in vertical conduits, and thereby prevent recurrence.

Conduits that exceeded the support requirements of General Construction Specification G-38 were analyzed and conduit support points with bearing pressure greater than allowable were inspected and supports added as required.

Cable Support in Vertical Trays

TVA's construction specification requires that cables in vertical trays be supported in accordance with the National Electric Code to prevent long term cable damage. The installation specification stated that this support may be provided by tie wraps. However, TVA had no basis to verify that cable ties can provide adequate support.

TVA evaluated the acceptability of various tie wrap configurations as support systems. If a configuration was found to be inadequate, it was shown by analysis, similarity to other installations, or testing that no cable damage had occurred or would occur. Cable support was added for those cables in which the cable bearing pressure, conductor and insulation strength, or the resultant loading imparted to the cable terminations or splices exceed manufacturers' limits, as a result of the cable weight. To prevent recurrence, TVA revised the cable installation specification and site procedures to identify acceptable methods for support of cables in vertical trays if the existing support system proved to be inadequate.

Cable Proximity to Hot Pipes

Cables were designed for 40 year life at ambient temperatures that do not include the local effects of hot pipes which increase local temperature and can degrade the cable insulation and shorten the life of the cables. For Unit 1, criteria were developed to detail required clearances between cable/raceways and hot pipes/valves to eliminate this potential impact. Class 1E cables were walked down against the criteria to ensure that adequate separation existed between the cables and hot pipes/valves. Deviations were resolved by analysis, change of pipe insulation or raceway rework.

Cable Pullbys

A 1989 inspection of electrical cables found cable insulation damage. This damage resulted in exposure of five instrumentation cables in the Unit 2 Reactor Protection System. Laboratory analysis confirmed TVA's initial assessment that the damage occurred as result of cable pullby. During the scope assessment effort, additional cables were removed and additional damage was found. These deficiencies were addressed at the time.

For Unit 1, TVA identified those locations where cable pull tension and cable side wall bearing pressure had exceeded certain safe threshold values, and cables that were most susceptible to this damage mechanism based on the conduit configuration. Cables that were in high risk and medium risk conduits were replaced. The threshold between low and high risk categories was validated via hi-pot testing or visual inspection and cables in the low risk category conduits were accepted-as-is based on the hi-pot tests performed on a sample of low risk category conduits.

Cable Bend Radius

TVA determined that the minimum bend radius recommended by the Insulated Cable Engineers Association had been violated at WBN. Excessive bending has the potential of damaging cables and adversely affecting their performance.

To resolve this issue on Unit 1, TVA established bend radius parameters (upper and lower bounds) for class 1E cables and revised General Construction Specification G-38 to include the bend radius requirements for cable installation. Cable was then categorized based on 10CFR50.49 requirements, classification and voltage level. Cables were then inspected and replaced, retrained or their qualified life was reduced, based on bending or kinking relative to upper and lower bound bend radii.

Cable Splices

TVA's internal review of WBN splicing details and experience at SQN indicated that the installed splices may not conform to the qualified configurations and materials tested by the vendor. To resolve this issue a list of Class 1E cable splices in harsh and mild environments was developed. Cables and splices were identified by reviewing equipment qualification binders and construction records to determine which equipment uses pigtails for field cable connection. 10CFR50.49 harsh environment cable splices were replaced and some mild environment cable splices were reworked. A sampling program was implemented to verify that the splice list was complete for intermediate splices.

Cable Sidewall Bearing Pressure

At WBN, sidewall bearing pressure (SWBP) was not properly addressed in the design and installation process and installations may have exceeded the allowable value. To resolve this issue on Unit 1, TVA conducted a walk down to identify worst case conduit configurations, calculated the expected pulling tension and SWBP for those worst case conduits and performed a test to determine increased allowable SWBP values, based on actual cables used at TVA nuclear plants. These results were reviewed by a third party contractor.

TVA revised construction specifications to require that SWBP be limited to the values determined by the above activities. WBN site installation procedures were revised to provide explicit cable SWBP restriction to cable pulling limits.

Analysis of the severe case conduits against these limits revealed that the cable in one conduit may have exceeded these values and this cable was replaced. An additional sample of conduits, all in harsh environment, was examined and none exceeded allowable SWBP.

Pulling Cable through 90° Condulet and Flexible Conduits

A concern was raised for the potential damage to cables in 90° condulets due to the small supporting surface the inside corners of condulets provide for cables under tension. Those corners can in time cut into the insulation, or the conductor can creep through the insulation, reducing the insulation level of the cables. Also, there was a concern at WBN that when cable is pulled through a flexible conduit segment in a bend, in the middle of a conduit run, it can be subjected to very high frictional forces that can tear the cable jacket and insulation.

TVA evaluated cables pulled through mid-route flexible conduits which had been tested for pullby damage, and inspected cables removed and confirmed that no damage was caused by the mid-route flexible conduits.

Computerized Cable Routing System (CCRS) Software and Database Verification and Validation

CCRS was used to document information regarding cable routing. The information includes cable route in trays and conduits, cable type, cable weight, cable splices, circuit function and separation etc. There were concerns for the adequacy of CCRS. CCRS has been replaced by new software called ICRDS. The CCRS data has been transferred to ICRDS for Units 1 and 2.

2. Cable Tray and Cable Tray Supports

A combination of Employee Concerns, CAQs and NRC violations identified deficiencies with cable trays and their supports. The specific deficiencies included inadequate tray connections, inconsistencies between as-designed versus as-built tray configurations and their orientation, and failure to evaluate all loading on cable tray members. Programmatically, these issues were categorized as a lack of documented design qualification for certain cable tray hardware, installed configuration not complying with design output documents, and a lack of documentation to verify previous re-inspection.

The CAP for Unit 1 assured the structural adequacy and compliance with design criteria and licensing requirements by:

- Review and revision of design criteria.
- Review or development of design output requirements to comply with design criteria and to adequately translate TVA design requirements. Examples were development of generic validation calculations for typical hardware configurations and evaluation of critical cases.
- Walk down of field configurations to identify deviations from design output.
- Modifications to field conditions, where necessary, to ensure that they were consistent with design output documents.

The CAP also put improvements in place to ensure the adequacy of new or modified cable trays and their supports. This was accomplished by training to revised design criteria and strengthened Nuclear Engineering (NE) procedures to allow plant configuration to be changed only on the basis of NE approved drawings, and by adding inspection requirements to verify fittings and connectors are installed consistent with Design Basis Document requirements.

3. Design Baseline and Verification Program

TVA became aware of inconsistencies in WBN licensing and design basis documentation as well as plant configuration issues as a result of several internal and external reviews that were conducted both at the plant and at the corporate level. The following conditions were identified by these reviews:

- Inconsistencies between the WBN Final Safety Analysis Report and WBN design documentation
- Incomplete and some inconsistent design input information
- Missing, incomplete and out-of-date design calculations
- Inconsistencies between the actual plant configuration and the as-constructed drawings.

The causes of these conditions were found to be:

- Lack of effective licensing and design change control procedures and data bases to ensure that design requirements were maintained consistent with the FSAR and other commitments to NRC.
- Insufficient definition of design criteria and system description information at the level of detail needed to control design changes.

- Lack of a complete calculation listing to establish the full scope of calculations needed for WBN and procedures to ensure the calculations are maintained consistent with the WBN design.
- Lack of an effective definition of drawings to be maintained under configuration control and an ineffective system for keeping appropriate drawings "as-constructed" as plant changes are made.

The underlying root cause of this situation was determined to be ineffective design and configuration control measures.

TVA developed the WBN Design Baseline and Verification Program (DBVP) to correct the situation that had developed and to prevent the recurrence of such a situation by eliminating the root cause. The DBVP had four major components, each having objectives that addressed one or more of the above problems. These components and objectives of each were:

- Licensing Verification
  - Assure that commitments to NRC are captured in the appropriate highest level controlling document
  - Establish procedures to maintain compatibility between commitments and controlling documents.

- Design Basis
  - Establish system and topical design basis documents (DBD) that contain or reference appropriate engineering requirements including design basis commitments
  - Establish procedures to maintain the design basis consistent with changes to the plant, technical requirements and licensing commitments.

- Calculations
  - Assure the existence and retrievability of calculations that are technically adequate and consistent with the safety-related plant design
  - Establish a process for statusing calculations that will maintain them current with plant design changes.

- Configuration Control
  - Develop and implement an improved design change control system
  - Establish a single set of configuration control drawings (CCDs)
  - Utilize walk downs, evaluations or testing to verify that the functional configurations of the portions of plant systems that mitigate plant design basis events are consistent between CCDs and the constructed plant.

4. Conduit Supports

Employee Concerns, CAQs and the Weld Task Group identified deficiencies in the conduit support program. Subsequently, a sample of conduit supports was reviewed to gain insight into specific nonconforming attributes. A number of specific structural deficiencies were identified including inadequate conduit clamps, conduit runs supported at only one location, and excessively cantilevered conduit. While these specific discrepancies were different from those identified for cable trays and their supports, programmatically the problems were similar and fell into four major categories:

- Design Basis discrepancies.
- Design output not enveloping all design parameters.
- Installed configurations not in compliance with design documents.
- Discrepancies between as-installed configurations and inspection documentation.

The CAP for Unit 1 assured the structural adequacy and compliance with design criteria and licensing requirements by:

- Revisions to design criteria
- Updated design output documents including specifications to factor in changes to design criteria, changes to typical support details and new support details. As with cable trays, critical case evaluations were performed for those configurations that did not meet design criteria.
- Walk downs first to support critical case evaluations, then to identify configurations not enveloped by critical cases
- Modifications, as required
- Revisions of implementing procedures to ensure the adequacy of new or modified supports

5. Electrical Issues

The Electrical Issues CAP was initiated based on various employee concerns, conditions adverse to quality documents, and NRC findings related to electrical installation, materials and equipments. The root cause of these concerns was primarily the absence or incompleteness of specific guidelines in the development of design input or output documents and, in some instances, the lack of procedural details for the installation of electrical components.

The CAP addressed the adequacy of safety-related electrical installations in the following areas:

1. Flexible Conduit Installations
2. Physical cable separation and electrical isolation
3. Contact and coil rating of electrical devices
4. Torque switch and overload relay bypass capability for active safety related valves
5. Adhesive backed cable support mounts (ABSCM)

Each of these issues is discussed individually below.

Flexible Conduit Installations

The problems identified with flexible conduits were:

- Inadequate length to account for seismic/thermal movement
- Lack of compliance with minimum bend radius requirements
- Loose Fittings

To resolve these issues for Unit 1, TVA revised design output documents to more specifically define flexible conduit requirements for:

- Seismic/thermal movement
- Minimum bend radius
- Tightness of fittings

A list of flexible conduits attached to Class 1E pipe mounted devices was then developed to identify those flexible conduits which would experience both seismic and thermal movement. Finally, TVA walked down Class 1E flexible conduits and reworked those found to be damaged or in noncompliance with the design output documents.

Physical Cable Separation and Electrical Isolation

CAQs and an employee concern identified isolated cases of less than the minimum required separation as specified by IEEE 279-1971, Standard Criteria for Protection Systems for Nuclear Generating Stations, IEEE 308-1971, Standard Criteria for Class 1E Electrical Systems for Nuclear Power Plants and Regulatory Guide (RG) 1.6, Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems.

For Unit 1, this issue was subdivided into three issues and each was resolved separately. The issues were:

- Separation between redundant divisions of Class 1E raceways,
- Internal panel separation between redundant divisions of Class 1E cables,

- Coil-to-contact and contact-to-contact isolation between Class 1E and non-Class 1E circuits.

For inadequate separation between redundant divisions of Class 1E raceways, the raceways were reworked to meet the minimum separation requirement and site implementing procedures were revised to require specific signoffs for raceway separation attributes.

For inadequate internal panel separation between redundant divisions of Class 1E cables design criteria were revised to include more detailed requirements for internal panel cables, an engineering output document was issued to define the requirements for internal panel cable separation and a list of panels with redundant divisions of Class 1E cables was developed. Panels containing cables of redundant divisions were walked down to identify cables which did not comply with the revised engineering output document and these were evaluated to determine acceptability or reworked to meet required separation distances.

For coil-to-contact and contact-to-contact isolation between Class 1E and non-Class 1E circuits a calculation was developed to determine when coil-to-contact and contact-to contact isolation were acceptable, design criteria were revised to specify acceptable isolation methods and the existing Class 1E coil and contact devices used as isolators were reviewed to determine that they were qualified for their intended use.

Contact and Coil Rating of Electrical Devices

Problem Identification Reports were issued at WBN for deficiencies where the design and procurements of inductive devices contained in the circuits did not consider the inductive load ratings of contacts or the maximum credible voltage available at the device terminals.

To resolve this for Unit 1, TVA reviewed devices that performed inductive load switching and determined if the contacts had acceptable current ratings and reviewed inductive devices to determine if coils were qualified for the highest and lowest credible voltages. If a device could not be qualified, design output documents were issued to require replacement and qualified devices were installed.

Torque Switch and Overload Relay Bypass Capability for Active Safety Related Valves

In order to meet the intent of Regulatory Guide 1.106, Thermal Overload Protection for Electric Motors on Motor Operated Valves, certain active safety related valves required to operate during a design basis event must have thermal overload and torque switches bypassed to ensure operability. Employee concerns and CAQs identified that TVA did not provide thermal overload and torque switch bypass capability for certain active safety-related valves.

For Unit 1, TVA issued design criteria to provide the basis for determining which active valves were required to have their thermal overload relays and torque switches bypassed and issued a calculation to identify these values. System design criteria or system descriptions were revised to identify which valves within a system require thermal overload and torque switches bypass capability, design output documents were revised to provide the required thermal overload and torque switch bypass capability, and thermal overload and torque switch bypasses were installed where they did not already exist.

Adhesive Back Cable Support Mounts

A CAQ documented that vendors and TVA used Adhesive Back Cable Support Mounts (ABSCM) inside equipment to support and restrain wire and field cables in a neat and orderly fashion. The ABSCMs sometimes separated from the inside of the equipment and, as a result, may not have properly secured the wire or cable.

For Unit 1, TVA contacted the vendors of the panels/equipment to ascertain the technical requirements for the ABSCMs for the vendor's wiring, evaluated the use of ABSCMs for field wiring and issued a calculation identifying the technical requirements for existing ABSCMs. TVA then evaluated the as-installed conditions to determine if any corrective action was required, issued and implemented design output documents in the field and revised site implementing procedures to incorporate the necessary installation requirements and to restrict the use of ABSCMs.

6. Equipment Seismic Qualification

TVA internal reviews and employee concerns involving seismic qualification of equipment identified a number of deficiencies which were documented in CAQRs, PIRs and SCRs. These deficiencies involved configuration and document control issues and included specific technical discrepancies. Some of the more significant technical discrepancies were:

- Set B spectra accelerations (site specific with corrections) were greater than Set A (Modified Newmark, originally used) for equipment in Category I structures
- Inadequate qualification of anchorages for tanks, pumps, heat exchangers and equipment mounted on flexible supports
- Discrepancies associated with installation of solenoid valves

- Anchor bolt spacing/proximity violations.

To address these and other specific issues, TVA implemented the Equipment Seismic Qualification (ESQ) CAP. The objectives of the program were to provide assurance that Category I and I (L) equipment are seismically qualified, that qualification documentation is retrievable and this documentation is consistent with the design and licensing basis. The main elements of the program were:

- Review and revision of design bases to ensure that they were technically adequate and consistent interfaces existed between them and other design bases (e.g. for piping loads and documents that establish seismic classification)
- Resolution of specific technical issues utilizing the following tasks:
  - Document retrieval
  - Walk downs to identify and describe as-built vs. as-designed discrepancies and aid in the determination of the actions required to resolve them
  - Engineering evaluations, including bounding calculations, to qualify equipment for their as-built configuration, and modifications when equipment could not be qualified in the as-built configuration
- Development and population of the ESQ database to maintain and control data required to qualify equipment
- Process improvements to ensure appropriate interfaces and prevent recurrence.

7. Fire Protection

The Fire Protection (FP) CAP was initiated after the identification of a number of issues at TVAN plants to ensure that the WBN fire protection program was complete and in compliance with licensing requirements. The specific items that were among these issues included:

- In 1987, TVA identified fire-rated walls that were breached by HVAC ducts without fire dampers. The condition violated Appendix R requirements pertaining to fire rated walls that separate safety-related equipment of redundant trains.
- In 1988, several SQN Appendix R discrepancies were identified by TVA in response to employee concerns and by NRC during inspections of the SQN Fire Protection Program. TVA committed to perform a review of these issues for applicability to WBN.
- Deficiencies were identified with the Safe Shutdown Analysis (SSA) in 1989.

In response to the above issues and other more specific deficiencies, the Unit 1 FP Program (for Unit 1 and common areas) contained the following actions:

- TVA initiated a CAQR to document the measures taken to evaluate violation of the Appendix R requirements and DCNs were issued to correct the deficiencies.
- TVA reviewed SQN Appendix R concerns, as well as issues raised by the NRC during SQN inspections, for applicability to WBN. CAQs were initiated to address the SQN inspection report items which were deemed applicable to WBN and DCNs were issued to correct the deficiencies.
- A Fire Protection Compliance Review was performed to ensure WBN conformance with NRC requirements and applicable guidelines. The following typical areas of fire protection requirements were addressed:

  - Safe Shutdown Analysis (SSA)
  - Area Heat-up Analysis
  - Fire Hazards Analysis
  - Lighting and Communication
  - Post-Fire procedures
  - Associated Circuits
  - Modification Compliance Review
  - Fire Protection Training/Administrative Procedures

  The results of the Compliance Review were used as the basis for developing the remaining scope of work (calculations/analysis, DCNs and document updates) and the consolidation of fire protection documentation into an organized package to support and substantiate the Compliance Review.

- TVA updated the SSA based on the latest as-constructed plant configuration. Also, the lessons learned from the SQN and BFN Appendix R inspections were factored into the revised SSA.

8. Hanger and Analysis Update Program (HAAUP)

During the design and construction of WBN, NRC issued several Bulletins concerning piping analysis and pipe support design. At the same time, at Watts Bar, piping and pipe support deficiencies were being identified through employee concerns, CAQRs, NCRs, problem identification reports, significant condition reports, and internal/external audits and reviews. The identified issues are listed below with their root causes:

- Control of Design Input/Output
  - Design input was not consistently defined and controlled.

  - Design output was not clearly defined and, thus, was not consistently implemented by Construction.
- Design/Analysis Methodology
  - Design criteria for piping analysis and pipe support design did not specify a consistent and comprehensive set of design/analysis methods. In some cases relevant industry issues were not considered.
- Level of Design Documentation
  - Requirements for closure of unverified assumptions and documentation of engineering judgments were neither fully defined nor procedurally controlled.

The scope of the HAAUP activities for Unit 1 included Seismic Category I piping, Seismic Category I (L) piping and those instrument lines that could not be decoupled from their process piping, and associated supports. (Those instrument lines that could be decoupled were addressed in the Instrument Line CAP). The HAAUP also addressed the issue of pipe support component substitution for catalog items. The following corrective actions were taken to address the deficiencies:

- Criteria and procedures governing piping system analysis and design were reviewed to ensure that they were in compliance with industry practices and were technically sound. The results of this review were incorporated into the implementing criteria and procedures utilized for the HAAUP.
- As-built information for installed piping and associated pipe supports was obtained by walk downs using the updated procedures.
- Pipe stress and support calculations were updated or newly generated:
  - To incorporate changes in the seismic response spectra input to envelope sets B and C, and to add consideration of mass participation above 33 Hz.
  - To qualify as-built conditions in drawings and calculations.
  - To ensure drawings and calculations are in compliance with current design criteria and procedures.
- Resolution of open items and identified deficiencies were incorporated into the design documents.
- Design documents were updated to incorporate the as-built piping and support configurations.
- Modifications were performed, as required.
- Shop made pipe support components were evaluated using a sampling of the different types of components and a 95% confidence level that 95% of the supports satisfied design basis was used to establish acceptability.

9. Heat Code Traceability

Traceability issues were identified and documented in 1985 during a review of ASME Section III N-5 data packages. Subsequently, similar issues were identified through employee concerns and CAQRs. These concerns involved ASME loose piping and fitting material and ASTM material installed as welded attachments on ASME piping systems and were categorized as:

- ASME Class 1 systems that may contain ASME Class 2, Class 3 and/or ASTM piping for which adequate NDE may not have been performed
- ASME Class 2 systems that may contain class 3 piping, and ASME Class 2 and Class 3 systems that may contain ASTM piping for which adequate NDE may not have been performed
- ASME systems that may have ASTM plate material attached (welded)

For the Unit 1 program, which included common systems, the following corrective actions were taken:

- Accuracy of the information contained in the Heat Code Database (HCDB) was first verified by comparing it to information on CMTRs. This information was used to flag situations where the same ASME material (same heat code) was used in systems of different classifications.
- For Class 1 piping, surface NDE was performed on piping materials where the heat number was the same as for material used in a non Class 1 system. When NDE was not feasible, alternate analysis prescribed by the ASME Code was performed. Material which could not be examined or technically justified was replaced.
- For Class 2 and 3 piping, required NDE was performed for piping components where classification traceability was questionable and they were installed in locations where stress ratios exceeded 0.80 for welded carbon steel and 0.85 for welded stainless steel. Additionally, for cases where ASTM, ASME Section II and ASME Section III material which may have been upgraded to ASME Section III, Class 2 or 3 materials, the items were re-verified as meeting all other requirements of Section III (material specifications, QA and design) on a sampling basis. Engineering evaluations were performed on non-complying items to provide a basis of acceptance. Material determined to be unacceptable, was replaced.
- ASTM plate attachment material used in ASME applications was determined to be acceptable by verifying equivalence to an ASME specification, that it was supplied to an acceptable QA program and

that it had the necessary NDE performed. Material that could not be verified or justified as being acceptable was replaced.

- Recurrence control included revising the General Construction Specification to include specific ASME requirements for reclassification of material and site implementing procedures to require CMTR traceability of materials to be installed.

10. HVAC Duct and Duct Supports

The HVAC Duct and Duct Supports CAP was implemented to address adverse conditions, which can be characterized as: incomplete design basis; inadequate design documents; as-built configurations not in conformance with existing design documents; inadequate or incomplete inspection documentation; and incomplete instructions.

For Unit 1, TVA resolved these issues via the following four tasks:

- Completing the design basis by reviewing and revising the design criteria, issuing supporting calculations as necessary and updating the FSAR to be consistent with the upgraded design criteria,
- Updating design output documents to be consistent with the completed design basis,
- Revising construction, maintenance and QA procedures to incorporate design output documents, and
- Developing bounding critical cases of existing installations and evaluating their adequacy, and performing unique evaluations or modifying installations when they could not be qualified by the critical case evaluations, thereby ensuring compliance with design basis.

11. Instrument Sensing Lines

As a result of WBN employee concerns and conditions adverse to quality (CAQ) documents, an overall review of instrumentation related issues was initiated in 1985. The review determined that the technical issues associated with instrument sensing lines required a separate and independent program to evaluate the functional and structural problems on these lines. The program was labeled the WBN Instrument Line CAP. The problems identified fell into two categories:

- Functional problems related to instrument line slope.

  The scope of the functional issue (slope) included sense lines associated with instruments that perform a safety-related function. It also included sense lines associated with instruments that have a particular sensitivity to the effects of entrapped air.

A number of the sense lines were found to not conform to the minimum slope requirements specified on design output drawings. The number of lines involved and the lack of adequate configuration control for these lines resulted in the following actions to address slope requirements:

- Preparation of an Engineering Requirements (ER) Specification
- The identification of sense lines to be reworked to meet the ER
- Preparation and issuance of sense line isometric drawings, support drawings, calculations, and analysis
- Installation and inspection of the sense lines per design output requirements.

In addition to the development of the ER Specification, other recurrence control measures were implemented for future design, installation, and inspection activities. These included site implementing procedures to incorporate ER in the process for the installation, maintenance, and inspection.

- Structural problems related to:

  - Thermal effects
  - Pipe and tube bending devices
  - Compression fittings
  - Installation discrepancies (inadequate or loss of documentation of sense line supports).

  The scope of the structural issues included Seismic Category I and I (L) instrument lines, and their associated supports, which are analytically decoupled from the process lines. Responses to these issues are described below:

  Thermal Effects

  The review determined that instrument lines and associated supports were not designed to consider the effects of thermal expansion and operating modes for these lines indicated that portions of the sampling and radiation monitoring systems will operate at temperatures for which thermal effects could be significant. These Unit 1 lines were field sketched to identify material, line configuration and support type. The lines were analyzed for dead weight, seismic and thermal effects. Detailed line isometric drawings were prepared showing required line configuration, support type and material. Any deficiencies were corrected through the issuance of a design change package.

Recurrence control was ensured by the issuance of a procedure which identified coordination requirements between Engineering disciplines associated with the qualification of thermal instrument lines.

Pipe and Tube Bending Devices

It was also determined that site implementing procedures used to qualify pipe and tube bending devices were not rigorously executed and that qualification records for the bending were not always maintained.

The corrective action consisted first of a sampling program which considered 200 randomly selected bends from an estimated total population of approximately 15,000 bends. The following attributes were evaluated: wall thickness reduction, ovality, acceptable bend contour, and surface condition. These samples were evaluated and found to be acceptable and bender qualification records were updated to incorporate the results of the sample program. Secondly, recurrence controls were established by the issuance of the Pipe and Tube Bending procedure and training requirements for personnel performing bending.

Compression Fittings

Various compression fitting installations were found that were not in accordance with the fitting manufacturer's installation requirements. A sample inspection of 107 compression fittings used on instrument lines was performed and 60 discrepancies were identified. The discrepant installations were categorized as: tubing cuts that were not deburred, tubing that was not bottomed out inside the fittings, nuts that were not properly tightened, and ferrules that were either judged to be unidentifiable, missing, or reversed. Also, certain discrepant fitting installations included parts supplied by different manufacturers.

TVA performed a vibration and pressure test program for the identified discrepant compression fitting installations. This program included testing of the effects of tubing burrs on flow rate as well as testing of the integrity of fittings with various installation deficiencies by tensile pullout, vibration and seismic tests. The results of these tests demonstrated that for the instances where tube ends were not deburred, tubes were not bottomed out, or nuts were not properly tightened, fitting performance was still satisfactory. Also, normal operation vibration testing did not result in leaks in any of the samples tested and seismic testing only produced very slight leakage (undetectable on the pressure gauges) in 2 of the 47 samples. The seismic tests were conservative and represented a severe test of fitting integrity.

The test program for fittings with missing, reversed, or unidentified ferrules determined that:

- Missing ferrules would cause a definite leak during pressure testing.
- Reversed ferrules would leak if they are "CPI" fittings.
- Reversed ferrules would not leak if they are reversed "Hi-Seal" ferrules.

TVA determined that for these three particular types of questionable ferrule installations, unacceptable installations would be detected during pressure testing due to leakage. If these questionable fitting configurations are used in instrument lines that are not pressure tested, there would be no driving force to create any significant leakage.

Therefore, the corrective action consisted of the following activities:

- Instrument lines designated as Seismic Category I or I(L) were pressure tested in accordance with appropriate piping code requirements
- Fittings seeing radioactive service in lines not pressure tested (i.e., drains) were re-inspected to verify installation in accordance with manufacturer's recommendations, and discrepancies found during the inspection were repaired or replaced.

In summary, for ferrule installations, since pressure testing was performed as required and leaking compression fittings were repaired or replaced, the final configurations were ultimately acceptable.

Recurrence controls consisted of revising specifications, design drawings and procedures, and requiring personnel to be trained and certified before they could perform installation and inspection activities.

Installation Discrepancies

A condition was identified whereby some instrument line support documentation was determined to be lost or incorrect and it was not apparent that this condition did not apply to all Seismic Category I and I (L) instrument line supports. The corrective action consisted of a random sample of 60 instrument line supports selected for a detailed evaluation to determine the acceptability of the as-built condition. The evaluation determined that the instrument lines and supports would comply with existing design basis requirements provided all attachment clamps and bolts were properly installed. The supports were then walked down to assure that the proper clamps were used and their installation was in accordance with established engineering requirements and, when necessary, the supports were reworked.

Recurrence controls consisted of updating site implementing procedures to ensure adequate installation and inspection requirements were in place.

11. Prestart Test

The Prestart Testing Program CAP for Unit 1 was withdrawn with the resubmittal of Chapter 14 of the FSAR to conform to the requirements of Regulatory Guide 1.68. The entire program is described in Chapter 14 of Amendment 91 of the FSAR.

12. QA Records

A review of WBN construction, maintenance and operations quality records required for licensing found that some records:

- Were not retrievable in a timely manner or potentially missing
- Were not maintained in proper storage
- Had quality problems (were incomplete, technically or administratively deficient)

To address these issues the QA Records CAP was developed. The objectives of this CAP were to:

- Ensure adequate storage and retrievability of required WBN construction, maintenance and operations records.
- Resolve quality and technical problems related to WBN construction, maintenance and operations records.
- Ensure programs adequate to prevent reoccurrence of records problems are established.

During the course of implementation of the CAP additional records issues were identified. Evaluation of these issues indicated a need to expand the scope to address the full extent of condition by including a broader set of records categories. This was accomplished through an Additional Systematic Records Review (ASRR) of ANSI N45.2.9 Appendix A record types, which was incorporated into the CAP. This review involved both records and hardware and was based on sampling and statistical analysis. It provided a high level of confidence in the adequacy of QA Records.

14. Q-List

The problems associated with the WBN Q-List Program identified in the CAP included:

- Multiple Q-Lists
- Inadequate training
- Lack of and improper classifications
- Wrong component identification

The objectives of the Q-List CAP were to:

- Develop a new Q-List
- Compare this new Q-List to the old Q-List to identify upgraded components
- Review maintenance and modification activities performed since 1984 to assure that those activities had the appropriate QA program controls applied.

As part of corrective action for this CAP, over 5000 component classification upgrades were identified during the comparison of the new and old Q-Lists. No field work resulted from these upgraded components.

15. Replacement Items

Replacement Items CAP was established as a result of weaknesses in TVA's policies, procedures, and practices in the area of commercial grade items purchased for installation in safety-related applications. Previous TVA policies and procedures did not adequately direct and control engineering involvement in the procurement process used to purchase replacement items. Neither did the TVA procedures incorporate industry (EPRI and NUMARC) guidance or comply with NRC Generic Letters 89-02 and 91-05.

The CAP grouped the issues into four categories:

- Current and future purchases
- Current warehouse inventory
- Plant installed items from previous maintenance activity
- Replacement items installed by previous construction activities

To address these categories, TVA:

- Created the Procurement Engineering Group, dedicated to review and evaluation of each procurement made for safety-related applications and developed a process for these activities.
- Created the Material Improvement Project to evaluate the adequacy of current inventory with respect to technical adequacy, QA receipt inspection and material storage.
- Back checked materials installed from previous maintenance activities to ensure that a proper documentation trail existed from the warehouse to maintenance history for each item.
- Reviewed the construction group's procurements of replacement items. This review indicated that the required documentation for parts traceability was available and that the materials were procured properly with engineering involvement.

16. Seismic Analysis

An independent review of the seismic analysis calculations for seismic Category I structures performed as part of the DBVP, the ECP and CAQRs identified issues and concerns that required further evaluation of the calculations, licensing requirements and design criteria.

The issues identified were consolidated into the following categories:

- Integration time step used to perform time history analysis.
- Soil properties and soil-structure interaction concerns
- Torsional modeling of structures
- Seismic analysis criteria for the Additional Diesel Generator Building
- The effect of floor and wall flexibility on design of SSC (structures, systems and components) in Category I structures

The seismic design basis for WBN is the Modified Newmark design spectrum (known as Set A) anchored at 0.18 g horizontal and 0.12 g vertical for the Safe Shutdown Earthquake (SSE). The Operating Basis Earthquake (OBE) is equal to one-half the SSE. The design basis spectra were determined to be acceptable by comparison with the Site Specific Spectra developed in 1979. The seismic design basis documented in the WBN FSAR and reviewed by NRC was accepted and documented in the WBN SER. However, consideration of the above identified issues required a re-evaluation of the two spectra and this was done for Unit 1, with the appropriate determination of accelerations to apply to Unit 1 and common SCCs (Set B for site specific and Set C for the modified Newmark).

The Diesel Generator Building (DGB), Additional Diesel Generator Building (ADGB), Auxiliary-Control Building (A/CB), Condensate Demineralizer Waste Evaporator (CDWE) Building and the Intake

Pumping Station (IPS) are common and covered by Unit 1 scope and no additional work is required for Unit 2. However, for the Interior Concrete Structure (ICS), Shield Building (SB), Steel Containment Vessel (SCV), Refueling Water Storage Tank (RWST), North Steam Valve Room (NSVR) and Pipe Tunnels for Unit 2, the original design basis spectra need to be compared to the site specific spectra and the comparisons documented as required.

For commodities and components, the spectra comparison between analysis using the original spectra and analysis using site specific spectra will be performed under the separate CAPs for these items (HVAC, HAAUP, Conduit, Cable Trays, Integrated Interaction Program and Equipment Seismic Qualification).

17. Vendor Information

TVA conditions adverse to quality reports, employee concerns, and TVA and NRC audit findings identified problems with vendor information at Watts Bar. Specific problems identified included:

- Vendor information didn't match the plant configuration.
- Vendor information was inconsistent with associated TVA-developed design input/output documents.
- Vendor documents were incorrect or out of date.
- Vendor manuals were lost or were uncontrolled.

The Vendor Information CAP for Unit 1 addressed these problems and their causes via the following actions:

- Relevant vendor information for safety-related and quality related Unit 1, common and Unit 2 components needed for Unit 1 operation were identified, reviewed for technical adequacy and consolidated into applicable Vendor Technical Manuals and Vendor Technical Documents, which were issued as controlled documents.
- Standard Procedure SSP-2.10 (which is now superseded by the current TVA procedure SPP-2.5) was issued to control vendor manual update activities.
- Open Item Reports were generated, tracked, and controlled to resolve the inconsistencies found in the vendor documents.
- Vendor drawings which included information necessary to support safety-related plant activities, but were not in "Approved" status, were reviewed and approved.
- DCNs were issued to resolve identified design discrepancies/open items.

18. Welding

NRC inspections, NCR's, CAQR's, audit findings and employee concerns identified programmatic and implementation deficiencies associated with safety related welding activities. In response to these issues, TVA established the TVA Welding Project to perform a review and determine the adequacy of the overall welding program, including that at WBN. Subsequently, the Welding CAP was established to ensure that Unit 1 safety-related welds meet TVA licensing requirements and provide specific corrective actions to address the prior issues and those identified by the Welding Project. The Welding CAP included both installation and vendor welding deficiencies which were related to weld quality, inspections, NDE, fabrication/installation code compliance, and associated documentation.

For Unit 1, the Welding CAP consisted of three phases:

- A programmatic assessment
- An in depth review of the implementation of the welding program and corrective actions to address specific discrepancies
- Program enhancements to prevent recurrence

The specific deficiencies that had to be addressed for Unit 1 involved structural steel, piping components, pipe supports, instrument panels, HVAC ductwork and vendor supplied component such as tanks and heat exchangers. The types of deficiencies included:

- Designs that did not satisfy design criteria for welding
- Lack of documentation of required visual inspections
- Indications or weld discontinuities
- Radiographs accepted with rejectable indications, inadequate radiographic techniques, and radiographic film identification discrepancies
- Misinterpretation of the ASME Code
- Discrepancies on vendor performed welds
- Errors on installation documentation

These problems were addressed by a combination of techniques that included the following:

- Re-inspections to validate results and support analysis
- Conservative bounding analysis
- Evaluation of as-is condition to determine acceptability
- Repairs, if necessary

Special Programs

1. Concrete Quality

An employee concern identified 24 issues related to concrete that subsequently resulted in the identification of three significant conditions involving concrete quality. They were:

- Some concrete mixes did not meet design compressive strength requirements,
- The use of mortar was not properly controlled, and
- Concrete sampling frequencies did not always comply with the requirements identified in specifications.

These issues were resolved and closed for Units 1 and 2 by the Concrete Quality Special Program. An NRC inspection (Inspection Report 390/90-26 and 391/90-26 – January 8, 1991) concluded that the concrete quality concern issues were resolved. Therefore, this program is complete for Unit 2.

2. Containment Cooling

During performance of EQ activities for Unit 1, TVA documented in a CAQR that the post-accident pressure and temperature analysis for the lower compartment in containment failed to consider the long-term effects of a main steam line break inside containment for a plant going to hot standby conditions as opposed to cold shutdown. 10CFR50.49 (e).1 requires that, "The time-dependent temperature and pressure at the location of electrical equipment important to safety must be established for the most severe design basis accident during or following which this equipment is required to remain functional." In order to ensure that this requirement is satisfied, TVA performed the Containment Cooling Special Program to develop time dependent temperature profiles for the lower compartment, which were then used for EQ. This was accomplished by the following tasks.

- Correcting the long-term containment temperature profile for the lower compartment considering the design basis Main Steam Line Break (MSLB) event using the Ice Condenser and Containment Spray systems as safety-grade systems for removing containment ambient (post accident) heat.
- Upgrading the Lower Compartment Cooler (LCC) units and associated ducting, with the exception of the LCC coils, to safety-grade, thus ensuring a qualified means of providing air circulation to sub compartments of the lower containment to prevent hot spots from forming.

- Evaluating containment coatings transport and replacing non qualified coatings to ensure that such coatings did not affect sump screen performance.
- Using the revised calculated MSLB temperature profile to qualify components in the lower containment that are important to safety.

3. Control Room Design Review

The Control Room Design Review (CRDR) program was developed in response to the NRC requirement established following the Three Mile Island Station accident that licensees and applicants conduct a CRDR to identify and correct human factor discrepancies in their control rooms. NUREG guidance for the conduct of the CRDR allowed licensees to perform a Preliminary Design Assessment (PDA) to identify any Human Engineering Discrepancies (HEDs) and establish a schedule for corrections with NRC staff approval and complete a full CRDR at a later date.

TVA performed a PDA and submitted the results to the NRC in January 1981. Discrepancies identified resulted in license commitments to implement corrective actions to resolve them and a CRDR Summary Report was identified as a license condition. TVA conducted a CRDR and submitted a CRDR Summary Report in October 1987. The CRDR addressed the man-machine interfaces and potential misapplication of human factor principles in the main control room, the auxiliary control room and the adjacent switch transfer rooms. Using a qualified multi-disciplined review team, TVA established a review program plan incorporating accepted human factor principles, gathered and reviewed required plant design information, surveyed the Control Room, identified and assessed HEDs, determined design improvements required and verified that the design improvements would address deficiencies and not create new ones.

Actions to ensure recurrence controls included:

- Corporate Engineering issuing Human Factor Design Guides
- Site Engineering issuing Human Factor Design Criteria
- The Design Change Process requiring human factors to be addressed

The CRDR Program was the subject of a series of audits and reviews, which led to revisions and further development of HED corrective actions for Unit 1, common equipment needed for Unit 1 and Unit 2 equipment needed to support Unit 1. The Unit 1 program was completed in 1995.

4. Environmental Qualification of Electrical Equipment

10 CFR 50.49 requires that equipment used to perform a necessary safety function is capable of maintaining functional operability under all service conditions postulated to occur during their installed life under normal operating and accident conditions. It is furthermore required that the evaluation of equipment qualification (EQ) be documented and maintained in auditable files. TVA conducted a management review of the EQ programs at SQN, BFN, and WBN in July and August of 1985. This review indicated that much of the qualification documentation was not fully auditable and, in some cases, the documentation available did not demonstrate full qualification.

The Equipment Qualification Special Program was initiated to address these discrepancies. Its primary objective was to document that safety related electrical equipment installed in the plant was qualified to perform their designated function in the environment to which they will be subjected during normal plant operation as well as during postulated accidents. Further, that programs and procedures have been established to ensure that qualification is maintained as future plant modifications are made. TVA put in place the tools to accomplish these objectives and, while they have evolved to satisfy today's requirements, these tools form the basis of the processes used to implement the WBN Unit 1 EQ Program, those currently used at TVA's nuclear plants and what will be implemented for Unit 2. These processes include:

- EQ program procedures which are the basis of plant procedures to maintain EQ over the operating life of the plant
- Consistent documentation requirements for the list of electrical equipment located in harsh environments and required to function after an accident, and the EQ Documentation Package providing documented evidence of the qualification of equipment for its specific application and environment
- Sources of information which provide the above evidence and appropriate reviews of this information
- Incorporation of EQ considerations into maintenance activities for equipment

The activities performed to satisfy the objectives for both EQ items qualified in place and those installed during the Unit 1 completion effort included:

- Analyses of the effects of pipe breaks (HELB and MELB) on temperature, humidity, dose and water level at various locations in containment and auxiliary buildings to establish the parameters for areas of the plant containing equipment that must meet 10CFR 50.49 requirements.

- Identification of 10CFR 50.49 equipment in these areas, the 50.49 list. This list included electrical equipment located in harsh environment and required to function after an accident. It was developed through a series of steps:

  - A systems analysis to determine for each DBA those equipment items required to ensure completion of a safety-related function.
  - For each item, a review of drawings to identify those ancillary devices and cable required to operate or maintain electrical integrity to ensure completion of the item's safety related function.
  - This list of items is reduced by performing a failure analysis which eliminates those components whose failure would not prevent achievement of the required safety action.

- Establishment of EQ binders that contain the qualification information in an auditable manner. This was accomplished using a documented process to direct completion of documented evidence of qualification of equipment for their specific application and environment. A package was developed for each Unit 1 equipment type. The package included:

  - Items comprising the equipment type
  - Checklist for evaluation of qualification
  - Analysis and justification of qualification
  - Qualification documents
  - Field verification data
  - Qualification Maintenance Data Sheets
  - Open items and deficiencies

5. Master Fuse List

Conditions adverse to quality documents and Nuclear Regulatory Commission findings identified deficiencies related to the design and configuration control of over current protection devices and the misapplication of Bussman KAZ actuators as protective devices. The programmatic issue was lack of control of these devices on the master fuse list and the lack of procedural guidance for the development of the Master Fuse List.

This Special Program was established for Unit 1 to resolve these deficiencies. The program included three primary elements:

- To address configuration control deficiencies, a baseline master fuse list was developed using as-designed schematics, connection drawings and vendor drawings to establish a comprehensive list of 1E fuses for Unit 1, and Unit 2 fuses needed to support the operation of Unit 1 systems; then walk downs were performed to gather as-installed information to be included on the list.

- To resolve the Bussman KAZ actuator misapplication a review was performed of schematic and connection drawings to identify KAZ locations and then an analysis performed and a DCN developed to replace the KAZ devices with conventional fuses. Unit 2 circuits that were needed for unit 1 operation were identified and redesigned to eliminate the KAZ actuators.
- To correct deficiencies involving redundancy provided to electrical penetration assemblies, an analysis was conducted to verify that redundant protection was provided by design and, when not the case, identified deficiencies were corrected.

While the principle focus of the program was on 1E Safety-Related equipment subsequent efforts have resulted in the program evolving to establish similar controls and practices for fuses needed to support the operation of the station.

6. Mechanical Equipment Qualification

The Mechanical Equipment Qualification (EQ) Program was initiated in response to an NRC requirement that TVA conduct a documented evaluation of the ability of safety-related mechanical equipment located in harsh environment to perform their intended functions, as required by GDC-4 of Appendix A of 10CFR50.

The Unit 1 program utilized existing temperature and dose conditions developed for electrical equipment to satisfy 10CFR50.49. The program then identified active safety-related mechanical equipment located in harsh environments; analyzed the non-metallic subcomponents for effect of thermal and radiation conditions; produced controlled binders to establish and maintain qualified status for life of plant; and issued DCNs to modify the plant consistent with qualification tests and analyses.

7. Microbiologically Induced Corrosion (MIC)

Due to leakage events in several water systems including Essential Raw Cooling Water and MIC degradation at other TVAN plants, TVA committed to a corporate program to address MIC in 1987. In addition, TVA committed to specific actions to address requirements of NRC Generic Letter 89-13, "Service Water System Problems Affecting Safety-related Equipment" and the potential for existing MIC conditions in Unit 1.

The special program for Unit 1 included:

- Identifying systems potentially affected by MIC via testing of water samples
- Performing visual inspections
- Reviewing design and operating documents and pre-existing NDE results to identify vulnerable locations
- Assessing MIC-infested locations identified during visual inspections
- Repairing unacceptable damage to Code requirements
- Installing improved biocide treatment and a long term chemical clean up system

This was later augmented by the implementation of SPP-9.7, Corrosion Control Program, which specifies the programmatic and organizational requirements for management of the MIC and Macrofouling Program and delineates the program's key elements.

8. Moderate Energy Line Break

Moderate energy lines are defined as piping that during normal conditions, are either in operation, or maintained pressurized (above atmospheric pressure) at temperatures less than or equal to 200 degrees F or pressures less than or equal to 275 psi. For these lines, TVA determined that documentation was inadequate to justify that there were no unacceptable consequences as a result of flooding in a Category I structure outside of containment following a moderate energy line break (MELB).

For Unit 1, TVA evaluated whether essential equipment and structures were either unaffected by postulated flooding due to a pipe break on a moderate energy pipe, or are designed, specified, and/or qualified for the environment caused by such flooding. The evaluation involved pipe break analyses, determination of postulated break locations, determination of postulated flooding levels, and equipment qualification evaluations. In those instances where it was determined that an item was impacted and it could not be qualified, modifications were performed to make the necessary plant upgrades. Modifications providing curbs, raising junction boxes and adding or removing weather stripping were performed.

9. Radiation Monitoring System

The WBN Radiation Monitoring System (RMS) Special Program was established to address deficiencies identified by employee concerns, CAQs and internal reviews. These deficiencies involved RMS design, documentation, installation, and hardware, and are categorized in three areas of concern. These are:

- Sample line deficiencies included excessive sample line length, incomplete heat tracing, minimum bend radius violations, incorrect slope, and violation of sample line separation requirements.

- Design and documentation deficiencies included:

  - Design defects on sample flow equipment
  - Purge capability following an accident not provided in the design of some radiation monitors
  - System interlocks with containment isolation were not provided in the containment upper and lower compartment monitor design
  - Undocumented modifications performed on the RMS rate meters
  - RMS rate meter cable damage due to rough, unfinished edge of the radiation monitors' cabinet.

- Calibration issues included inadequate documentation of the traceability of the primary calibration records to support equipment testing and calibration resulting in uncertainty in the validity of the equipment calibration and the level of radioactivity indicated or alarmed.

The actions to address these deficiencies for Unit 1 were to review and update the RMS design basis, including applicable requirements of Regulatory Guide 1.97, evaluate the RMS against this design basis, and to implement modifications to correct RMS deficiencies. This also included an evaluation of the RMS design, documentation, and installations against the updated design criteria to verify the acceptability of the installation or to identify required modifications for those monitors included in the technical specifications and modifications or reworking of existing documentation to correct identified documentation.

10. Soil Liquefaction

The potential for soil to liquefy is a design consideration at Watts Bar Nuclear Plant and the Soil Liquefaction Special Program was implemented to assure that employee identified issues regarding the design and construction of soil mitigating measures on the west side of the Intake Pumping Station were addressed. The actions taken as part of this program were an evaluation of the concerns in three areas:

- Alternate material used for backfill
- Incomplete excavation of potentially liquefiable sand.
- Leakage between the IPS and Trench B.

To address these issues for Unit 1, TVA conducted an investigation into the concerns. In addition, an independent investigation was performed by R. L. Cloud and Associates with review by the late H. B. Seed of

University of California, Berkeley (a noted expert in the area of soil liquefaction). Based on the investigations, it was concluded that the areas of concern did not have a detrimental effect on liquefaction mitigation measures.

Implementation of the Soil Liquefaction Program is complete for both units. The FSAR revisions resulting from the work were submitted to NRC as part of Amendment 63. The NRC documented their review and acceptance that the WBN underground barriers provide sufficient confinement for liquefied soil in the affected area in Safety Evaluation Report Supplement No. 3. The NRC endorsed the approach for the Soil Liquefaction Special Program by NUREG-1232, subject to inspection of TVA's actions. Final inspection accepting the completion of this program was reported in NRC Inspection Report 390/92-45 and 391/92-45 – February 17, 1993. There will be no impact on this status as a result of completion of Unit 2.

11. Use-As-Is CAQs

In 1986, Nuclear Engineering conducted an audit of the Watts Bar engineering program activities related to the handling of construction non-conformance reports. They identified that use-as-is and repair non-conformance dispositions were not reflected on drawings; there was inadequate justification for disposition of these types of non-conformances; and no project level procedural guidance was provided for use-as-is and repair dispositions. This program was initiated to address these issues.

To prevent recurrence, engineering procedures were issued to establish the requirements for handling CAQs including a specific requirement to ensure that appropriate design documents reflect the approved configuration for any use-as-is or repair disposition. These procedures also required the basis for approval of any use-as-is or repair dispositions be documented along with the disposition on the CAQ report.

For Unit 1, this was followed by the identification of CAQs that had a final disposition of either use-as-is or repair and performance of technical reviews of the latest revision of affected design documents considering the impact of the CAQ.

Attachment 3 - Listing of Generic Letters, Bulletins and TMI Action Items issued before 1995

Attachment 3 Listing of Rules, Bulletins and Generic Letters and TMI Action Items issued before 1995

1.	10CFR50.63 – Station Blackout
2.	IEB 76-02 – Relay Coil Failures – GE Type HFA, HGA, HKA, HMA Relays
3.	IEB 77-03 – On-Line Testing of the W Solid State Protection System
4.	IEB 79-02 – Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts
5.	IEB 79-14 – Seismic Analysis for As-Built Safety-Related Piping Systems
6.	IEB 79-21 – Temperature Effects on Level Measurements
7.	IEB 79-24 – Frozen Lines
8.	IEB 79-27 – Loss of Non-Class 1E I & C Power System Bus During Operation
9.	IEB 80-04 – Analysis of a PWR Main Steam Line Rupture with Continued Feedwater Addition
10.	IEB 80-05 – Vacuum Condition Resulting in Damage to Chemical Volume Control System Holdup Tanks
11.	IEB 80-06 – Engineered Safety Features Reset Control
12.	IEB 80-10 - Contamination of Non-radioactive System and Resulting Potential for Unmonitored, Uncontrolled Release of Radioactivity to Environment
13.	IEB 80-11 – Masonry Wall Design
14.	IEB 80-18 – Maintenance of Adequate Minimum Flow Thru Centrifugal Charging Pumps Following a Secondary Side High Energy Rupture
15.	IEB 80-20 – Failure of W Type W-2 Spring Return to Neutral Control Switches
16.	IEB 80-24 - Prevention of Damage Due to Water Leakage Inside Containment
17.	IEB 84-03 – Refueling Cavity Water Seal

18.	IEB 85-02 – Undervoltage Trip Attachment of W DB-50 Type Reactor Trip Breakers
19.	IEB 88-02 - Rapidly Propagating Fatigue Cracks in Steam Generator Tubes
20.	IEB 88-04 - Potential Safety-Related Pump Loss
21.	IEB 88-05 - Nonconforming Materials Supplied by Piping Supplies, Inc. and West Jersey Manufacturing Company
22.	IEB 88-08 - Thermal Stresses in Piping Connected to Reactor Cooling Systems
23.	IEB 88-09 - Thimble Tube Thinning in Westinghouse Reactors
24.	IEB 88-10 - Nonconforming Molded-Case Circuit Breakers
25.	IEB 88-11 - Pressurizer Surge Line Thermal Stratification
26.	IEB 89-01 - Failure of Westinghouse Steam Generator Tube Mechanical Plugs
27.	IEB 89-02 - Stress Corrosion Cracking of High-Hardness Type 410 Stainless Steel Preloaded Bolting in Anchor Darling Model S350W Swing Check Valves or Valves of Similar Nature
28.	IEB 90-01 – Loss of Fill-Oil in Transmitters Manufactured by Rosemount
29.	GL 78-02 –Asymmetric Loads Background and Revised Request for Additional Information
30.	GL 78-03 – Request for Information on Cavity Annulus Seal Ring
31.	GL 78-34 – Reactor Vessel Atypical Weld Material
32.	GL 79-20 – Cracking in Feedwater Lines
33.	GL 79-36 – Adequacy of Station Electric Distribution System Voltages
34.	GL 80-14 – LWR Primary Coolant System Pressure Isolation Valves
35.	GL 80-21 - Vacuum Condition Resulting in Damage to Chemical Volume Control System Holdup Tanks
36.	GL 80 -46/47 – A-12, Fracture Toughness and Additional Guidance on Potential for Low Fracture Toughness and Laminar Tearing on PWR Steam Generator Coolant Pump Supports

37.	GL 81-07 – Control of Heavy Loads (Addresses NUREG-0612)
38.	GL 81-21 – Natural Circulation Cooldown
39.	GL 82-28 - Inadequate Core Cooling Instrumentation System (Addresses TMI Item II.F.2)
40.	GL 82-33 – Supplement to NUREG-0737, Requirements for Emergency Response Capability
41.	GL 83-28 - Required Actions Based on Generic Implications of Salem ATWS Events: 2.2 – Equipment Classification and Vendor Interface (All SR Components) 4.1 – Reactor Trip System Reliability (Vendor Related Modifications)
42.	GL 85-02 - Recommended Actions Stemming From NRC Integrated Program for the Resolution of Unresolved Safety Issues Regarding Steam Generator Tube Integrity
43.	GL 85-12 – Implementation of TMI Item II.K.3.5 – Automatic Trip of Reactor Coolant Pumps
44.	GL 88-05 - Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR plants
45.	GL 88-11 – NRC Position on Radiation Embrittlement of Reactor Vessel Material and its Impact on Plant Operations
46.	GL 88-14 - Instrument Air Supply System Problems Affecting Safety-Related Equipment
47.	GL 88-17 - Loss of Decay Heat Removal
48.	GL 88-20 – Individual Plant Examination for Severe Accident Vulnerability
49.	GL 89-04 – Guidelines on Developing Acceptable Inservice Testing Programs
50.	GL 89-06 – TMI Action Plan Item I.D.2 – Safety Parameter Display System
51.	GL 89-08 - Erosion/Corrosion-Induced Pipe Wall Thinning
52.	GL 89-10 - Safety-Related Motor-Operated Valve Testing and Surveillance
53.	GL 89-13 – Service Water System Problems Affecting Safety Related Equipment

54.	GL 89-19 - Request for Actions Related to Resolution of Unresolved Safety Issue A-47 "Safety Implication of Control Systems in LWR Nuclear Power Plants" Pursuant to 10 CFR 50.54(f)
55.	GL 89-22 – Potential for Increased Roof Load Due to Changes in Maximum Precipitation
56.	GL 90-06 - Resolution of Generic Issues 70, "PORV and Block Valve Reliability," and 94, "Additional LTOP Protection for PWRs"
57.	GL 92-01 - Reactor Vessel Structural Integrity
58.	GL 92-08 - Thermo-Lag 330-1 Fire Barriers
59.	GL 93-04 - Rod Control System Failure and Withdrawal of Rod Control Cluster Assemblies, 10 CFR 50.54(f)
60.	NUREG 0612 – Control of Heavy Loads – Response is in 7/21/93 response to GL 81-07
61.	NUREG 0737 – TMI Items: I.C.1 Short Term Accident and Procedure Review I.C.7 NSSS Vendor Revision of Procedures I.C.8 Pilot Monitoring of Selected Emergency Procedures for NTOLs I.D.1 Control Room Design Review I.D.2 Plant-Safety-Parameter-Display Console II.B.1 Reactor-Coolant-Vent-System II.B.2 Plant Shielding II.B.3 Post-Accident Sampling System II.D.1 Relief and Safety Valve Test Requirements II.D.3 Valve Position Indication II.E.1.1 Auxiliary Feedwater System Evaluation, Modifications II.F.1.2.A Accident-Monitoring Instrumentation – Noble Gas II.F.1.2.B Accident-Monitoring Instrumentation – Iodine/Particulate Sampling II.F.1.2.C Accident-Monitoring Instrumentation – Containment High Range Monitoring

	II.F.1.2.D Accident-Monitoring Instrumentation – Containment Pressure II.F.1.2.F Accident-Monitoring Instrumentation – Containment Water Level II.F.1.2.D Accident-Monitoring Instrumentation – Containment Hydrogen II.F.2 Instrumentation for detection of inadequate core-cooling II.G.1 Power Supplies for Pressurizer Relief Valves, Block Valves and Level Indicators II.K.1.10 Operability Status II.K.3.1 Auto PORV Isolation II.K.3.3 Reporting SV/RV Failures/Challenges II.K.3.5 Auto Trip of RCPs II.K.3.9 PID Controller II.K.3.17 ECCS Outages II.K.3.25 Power on Pump Seals II.K.3.31 Plant Specific Analysis III.D.1.1 Primary Coolant Outside Containment III.D.3.3 In-Plant I2 Radiation Monitoring III.D.3.4 Control-Room Habitability.

Attachment 4 – List of new Regulatory Requirements and Generic Communications

Attachment 4 Listing of new Regulatory Requirements and Generic Communications

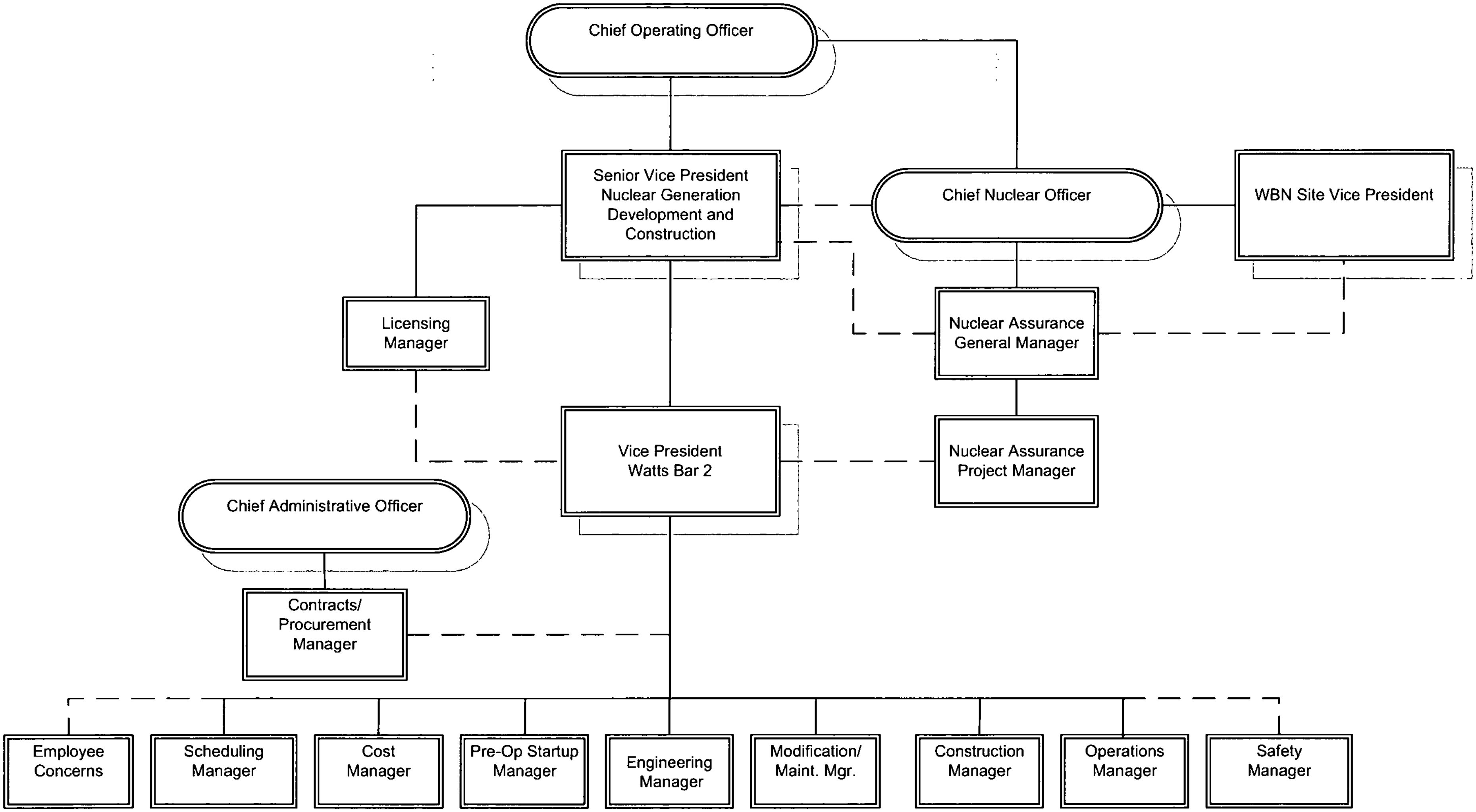
	Rule, Bulletin or GL	Title	Associated SRP Section
1.	50.65	Maintenance Rule	3.8.3, 3.8.4, 3.8.5, 5.4.2.2, 8.2, 8.3.1, 8.3.2
2.	50.68	Criticality Accident Requirements	9.1.1, 9.1.2, 12.3, 12.4
3.	50.60 and 50.61	Fracture Toughness Requirements -Pressurized Thermal Shock -Thermal Annealing	5.3.1. 5.3.2, 5.3.3
4.	Bulletin 96-01	Control Rod Insertion Problems (PWR)	
5.	Bulletin 96-02	Movement of Heavy Loads over Spent Fuel, over Fuel in the Reactor, or over Safety-Related Equipment	
6.	Bulletin 01-01	Circumferential Cracking of Reactor Pressure Vessel (RPV) Head Penetration Nozzles	
7.	Bulletin 02-01	RPV Head Degradation and Reactor Coolant Pressure Boundary Integrity	
8.	Bulletin 02-02	RPV Head and Vessel Head Penetration Nozzle Inspection Program	
9.	Bulletin 03-01	Potential Impact of Debris Blockage on Emergency Sump Recirculation	

	Rule, Bulletin or GL	Title	Associated SRP Section
10.	Bulletin 03-02	Leakage from RPV Lower Head Penetrations and Reactor Coolant Pressure Boundary Integrity	
11.	Bulletin 04-01	Inspection of Alloy 82/182/600 Materials Used in the Fabrication of Pressurizer Penetrations and Steam Space Piping Connections at PWRs	
12.	GL 95-03	Circumferential cracking of Steam Generator (SG) Tubes	
13.	GL 95-05	Voltage Based Repair Criteria for W SG Tubes Affected by Outside Diameter Stress Corrosion Cracking	
14.	GL 95-07	Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves	
15.	GL 96-01	Testing of Safety-Related Circuits	8.3.1
16.	GL 96-03	Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits	5.2.2
17.	GL 96-05	Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves	3.9.6
18.	GL 96-06	Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions	9.2.1, 9.2.2

	Rule, Bulletin or GL	Title	Associated SRP Section
19.	GL 97-01	Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations	4.5.2, 5.2.3
20.	GL 97-04	Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps	
21.	GL 97-05	SG Tube Inspection Techniques	
22.	GL 97-06	Degradation of SG Internals	
23.	GL 98-02	Loss of Reactor Coolant Inventory and Associated Potential for Loss of Emergency Mitigation Functions While in a Shutdown Condition	
24.	GL 98-04	Potential for Degradation of the ECCS and the Containment Spray System After a LOCA Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment	
25.	GL 03-01	Control Room Habitability	
26.	GL 04-01	Requirements for SG Tube Inspection	
27.	GL 04-02	Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at PWRs	
28.	GL 06-01	SG Tube Integrity and Associated Technical Specifications	
29.	GL 06-02	Grid Reliability and the Impact on Plant Risk and the Operability of Offsite Power	

	Rule, Bulletin or GL	Title	Associated SRP Section
30.	GL 06-03	Potentially Nonconforming Hemyc and MT Fire Barrier Configurations –	
31.	GL 07-01	Inaccessible or Underground Power Cable Failures that Disable Accident Mitigation Systems or Cause Plant Transients	8.3.1

Attachment 5 – Construction Completion Organization



Updated 7/25/07

Attachment 6 – Listing of Commitments Made in Letter

1. TVA will provide a regulatory framework submittal for WBN Unit 2 completion by January 31, 2008.

2. TVA plans to provide a red-line version of the WBN Unit 1 FSAR early in the project. The schedule for submitting this markup FSAR will be provided in the regulatory framework document.

3. Subsequent to the initial submittal, TVA intends to provide updates, as appropriate, to the regulatory framework submittal until the WBN Unit 2 commitments related to fuel load, startup and power operation are complete.

4. The WBN Unit 2 Preservice Inspection Program was last submitted to NRC on April 30, 1990. TVA will provide a revised program for NRC approval.

5. TVA will provide the Pressure Temperature Limits Report for WBN Unit 2 for NRC approval.