WESTINGHOUSE CLASS 3

WCAP-13470

WATTS BAR UNIT 1
TECHNICAL SPECIFICATIONS CRITERIA
APPLICATION REPORT

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SECTION 1

INTRODUCTION

1.1 BACKGROUND

This report documents the application of the Technical Specification screening criteria to the Watts Bar Unit 1 Technical Specifications (Draft, 1985). The criteria applied were established by the NRC Interim Policy Statement on Technical Specification Improvement (52FR3788); February 6, 1987. The criteria were provided for use in evaluation of Technical Specifications to determine whether a specification must be retained, or may be relocated to another licensee controlled document.

1.2 APPLICATION OF THE NRC SELECTION CRITERIA

The NRC Interim Policy Statement provided the following Technical Specification screening criteria:

CRITERION 1

Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

CRITERION 2

A process variable that is an initial condition of a Design Basis Accident (DBA) or Transient Analyses that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

CRITERION 3

A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

The criteria is applied to each Technical Specification Limiting Condition for Operation (LCO). If the Technical Specification meets any one of the above criteria, then the Technical Specification shall be retained in the new Watts Bar Unit 1 Technical Specifications based on the new Westinghouse Standard Technical Specifications, NUREG-1431.

PRA Criteria

If none of the criteria are satisfied, then the Technical Specification is a candidate for relocation and must be evaluated from a Probabilistic Risk Assessment (PRA) point of view. The PRA determines if the parameters, components, or systems addressed by the Technical Specifications have been modeled within the available literature on risk insights and PRA studies, and whether they are

of prime importance in limiting the likelihood or severity of accident sequences that are commonly found to dominate offsite health effects. Technical Specifications which meet this PRA criteria shall be retained in the Watts Bar Unit 1 Technical Specifications.

The NRC Interim Policy Statement criteria was originally applied to the Westinghouse Standard Technical Specifications (STS) NUREG-0452, draft Revision 5 and documented in WCAP-11618, Reference 1. The NRC documented the results of their review of WCAP-11618 in NRC Letter, Reference 2. Appendix B, Tables 1 and 2 of Reference 2 identified which W-STS specifications must be retained and which specifications may be relocated. These NRC determinations form the basis for the application of the criteria to the Watts Bar Unit 1 Technical Specifications. The Watts Bar Unit 1 Technical Specifications were compared to the W-STS requirements. The Watts Bar specifications with requirements comparable to the W-STS were dispositioned in accordance with the NRC Letter, Reference 2. Watts Bar specifications which were found to be unique, were evaluated in accordance with the NRC Interim Policy Statement criteria. If none of the criteria applied, then the unique specification was evaluated from a PRA point of view. The result of the PRA evaluation determined whether the unique specification was retained or relocated.

1.3 RESULTS

The recommended disposition of the 1985 Draft Watts Bar Unit 1 Technical Specification LCOs is presented in Table 1. These results are based on the application of the criteria and conclusions documented by the NRC. Those LCOs in the Watts Bar Unit 1 Technical Specifications which were not addressed in Reference 2, and therefore, not reviewed by the NRC; are indicated with "Not Reviewed" in the NRC Results column of Table 1. Supplementary information related to the basis of the information contained in Table 1, is provided as referenced notes at the end of the table.

Each LCO, which has not been previously reviewed by the NRC, has been subjected to the screening process. This process is documented on screening forms, which are included in Table 2. The screening forms furnish the justification for the retention of the specification or the relocation outside of the Technical Specifications. Each screening form, for LCOs which did not meet the screening criteria, references Appendix A, which documents the PRA evaluations performed to determine if the Technical Specification requirement is contained in a dominate risk sequence. If a requirement is not risk significant, and does not satisfy any of the NRC screening form criteria for retention, then it has been concluded that it may be relocated to another licensee controlled document.

Each LCO which is identified in Table 1 for relocation, or which will have specific requirements removed and relocated, such as cycle specific physics parameters (relocated to the Core Operating Limits Report), is listed in Table 3. The LCOs listed are organized by their intended destination.

Appendix A provides a detailed account of the analytical approach and assumptions used in the PRA evaluation. This Appendix also includes a description of the application of PRA to the Westinghouse MERITS program and

the available PRA studies for other similar plants. A justification is provided for the use of this material to assess the sensitivity of Watts Bar Unit 1 to plant incidents involving risk to the general public.

1.4 CONTROL OF RELOCATED TECHNICAL SPECIFICATIONS

The Technical Specifications that do not meet any of the Technical Specification screening criteria and are not significant risk contributors will be relocated outside of the Watts Bar Unit 1 Technical Specifications. Some of these Technical Specifications, for example, the Radiological Effluent Technical Specifications (RETS) will be relocated and controlled as part of the Offsite Dose Calculation Manual (ODCM) specified in the Administrative Controls section of Technical Specifications in accordance with NRC staff guidance provided in Generic Letters.

Watts Bar administrative procedures will be used for the relocated Technical Specifications which are not controlled as programs. These relocated specifications will be included in the Technical Requirements Manual (TRM). A copy of the TRM will be maintained with the Technical Specifications but as a separate document. Changes to the TRM will be controlled in accordance with the criteria of 10 CFR 50.59 and any other controlling regulation (e.g., 10 CFR 50.55a in the case of ISI/IST Programs) as described in the FSAR, Chapter 16.

1.5 REFERENCES

1. WCAP-11618, "Methodically Engineered, Restructured, and Improved Technical Specifications", MERITS Program - Phase II, Task 5, Criteria Application.

2. T. E. Murley to W. S. Wilgus, "NRC Staff Review of Nuclear Steam Supply System Vendor Owners ,Groups' Application of the Commission's Interim Policy Statement Criteria to Standard Technical Specification", May, 1988 (NRC Letter).

3. NRC Generic Letter 89-01, dated January 31, 1989, "Guidance for the Implementation of Programmatic Controls for RETS in the Administrative Controls Section of Technical Specifications and the Relocation of Procedural Details of Current RETS to the Offsite Dose Calculation Manual or Process Control Program".

4. NRC Generic Letter 88-16, dated October 4, 1988, "Removal of Cycle-Specific Parameter Limits from Technical Specifications."

5. NRC Generic Letter 88-12, dated August 2, 1988, "Removal of Fire Protection Requirements from Technical Specifications."

TABLE 1

WATTS BAR UNIT 1 SUMMARY LIST

Reactivity Control Systems

Unit 1 Number	STS-Rev 5 Number	Watts Bar Unit 1 Title (STS Rev. 5 for WB N/A)	NRC Results	Unit 1 Results	Notes
3.1.1.1	3.1.1.1	Shutdown Margin, $T_{avg} > 200°F$	Retain	Retain	1
3.1.1.2	3.1.1.2	Shutdown Margin, $T_{avg} \leq 200°F$	Retain	Retain	
3.1.1.3	3.1.1.3	Moderator Temperature Coefficient	Retain	Retain	2
3.1.1.4	3.1.1.4	Minimum Temperature for Criticality	Retain	Retain	
3.1.2.1	3.1.2.1	Flow Paths, Shutdown	Relocate	Relocate	
3.1.2.2	3.1.2.2	Flow Paths, Operating	Relocate	Relocate	
3.1.2.3	3.1.2.3	Charging Pump, Shutdown	Relocate	Relocate	
3.1.2.4	3.1.2.4	Charging Pumps, Operating	Relocate	Relocate	
3.1.2.5	3.1.2.5	Borated Water Sources - Shutdown	Relocate	Relocate	
3.1.2.6	3.1.2.6	Borated Water Sources - Operating	Relocate	Relocate	
3.1.3.1	3.1.3.1	Movable Control Assemblies, Group Height	Retain	Retain	2
3.1.3.2	3.1.3.2	Position Indicating Systems - Operating	Relocate	Retain	3
3.1.3.3	3.1.3.3	Position Indicating Systems - Shutdown	Relocate	Relocate	4
3.1.3.4	3.1.3.4	Rod Drop Time	Relocate	Relocate	4
3.1.3.5	3.1.3.5	Shutdown Rod Insertion Limit	Retain	Retain	2
3.1.3.6	3.1.3.6	Control Rod Insertion Limits	Retain	Retain	2

TABLE 1 (Continued)

WATTS BAR UNIT 1 SUMMARY LIST

Power Distribution Limits

Unit 1 Number	STS-Rev 5 Number	Watts Bar Unit 1 Title (STS Rev. 5 for WB N/A)	NRC Results	Unit 1 Results	Notes
3.2.1	3.2.1	Axial Flux Difference	Retain	Retain	2
3.2.2	3.2.2	Heat Flux Hot Channel Factor-$F_Q(Z)$	Retain	Retain	2
3.2.3	3.2.3	RCS Flow Rate and Nuclear Enthalpy Rise Hot Channel Factor	Retain	Retain	2
3.2.4	3.2.4	Quadrant Power Tilt Ratio	Retain	Retain	
3.2.5	3.2.5	DNB Parameters	Retain	Retain	

TABLE 1 (Continued)

WATTS BAR UNIT 1 SUMMARY

Instrumentation

Unit 1 Number	STS-Rev 5 Number	Watts Bar Unit 1 Title (STS Rev. 5 for WB N/A)	NRC Results	Unit 1 Results	Notes
3.3.1	3.3.1	Reactor Trip System Instrumentation	Retain	Retain	
3.3.2	3.3.2	Engineered Safety Features Actuation System Instrumentation	Retain	Retain	
3.3.3.1	3.3.3.1	Radiation Monitoring for Plant Operations	Retain	Retain	5
3.3.3.2	3.3.3.2	Movable Incore Detectors	Relocate	Relocate	
3.3.3.3	3.3.3.3	Seismic Instrumentation	Relocate	Relocate	
3.3.3.4	3.3.3.4	Meteorological Instrumentation	Relocate	Relocate	
3.3.3.5	3.3.3.5	Remote Shutdown Instrumentation	Retain	Retain	
3.3.3.6	3.3.3.6	Accident Monitoring Instrumentation	Retain	Retain	
N/A	3.3.3.7	Chlorine Detection System	Relocate	N/A	6
3.3.3.7	3.3.3.8	Fire Detection Instrumentation	Relocate	Relocate	7
3.3.3.8	3.3.3.10	Radioactive Liquid Effluent Monitoring Instrumentation	Relocate	Relocate	8
3.3.3.9	3.3.3.11	Radioactive Gaseous Effluent Monitoring Instrumentation	Relocate	Relocate	9
3.3.3.10	3.3.3.9	Loose-Part Detection System	Relocate	Relocate	
3.3.4	3.3.4	Turbine Overspeed Protection	Relocate	Relocate	10

TABLE 1 (Continued)

WATTS BAR UNIT 1 SUMMARY

Reactor Coolant System

Unit 1 Number	STS-Rev 5 Number	Watts Bar Unit 1 Title (STS Rev. 5 for WB N/A)	NRC Results	Unit 1 Results	Notes
3.4.1.1	3.4.1.1	Startup and Power Operation	Retain	Retain	
3.4.1.2	3.4.1.2	Hot Standby	Retain	Retain	
3.4.1.3	3.4.1.3	Hot Shutdown	Retain	Retain	
3.4.1.4.1	3.4.1.4.1	Cold Shutdown - Loops Filled	Retain	Retain	
3.4.1.4.2	3.4.1.4.2	Cold Shutdown - Loops Not Filled	Retain	Retain	
N/A	3.4.1.5	RCS Isolated Loop	Retain	N/A	6
N/A	3.4.1.6	RCS Isolated Loop Startup	Retain	N/A	6
3.4.2.1	3.4.2.1	Safety Valves - Shutdown	Relocate	Relocate	
3.4.2.2	3.4.2.2	Safety Valves - Operating	Retain	Retain	
3.4.3	3.4.3	Pressurizer	Retain	Retain	
3.4.4	3.4.4	Relief Valves	Retain	Retain	
3.4.5	3.4.5	Steam Generators	Relocate	Relocate	11
3.4.6.1	3.4.6.1	Leakage Detection Systems	Retain	Retain	
3.4.6.2	3.4.6.2	Operational Leakage	Retain	Retain	
3.4.7	3.4.7	Chemistry	Relocate	Relocate	
3.4.8	3.4.8	Specific Activity	Retain	Retain	
3.4.9.1	3.4.9.1	Pressure/Temperature Limits, Reactor Coolant System	Retain	Retain	12
3.4.9.2	3.4.9.2	Pressure/Temperature Limits, Pressurizer	Relocate	Relocate	
3.4.9.3	3.4.9.3	Overpressure Protection Systems	Retain	Retain	

TABLE 1 (Continued)

WATTS BAR UNIT 1 SUMMARY

Reactor Coolant System (Continued)

Unit 1 Number	STS-Rev 5 Number	Watts Bar Unit 1 Title (STS Rev. 5 for WB N/A)	NRC Results	Unit 1 Results	Notes
3.4.10	3.4.10	Structural Integrity	Relocate	Relocate	13
3.4.11	3.4.11	Reactor Coolant System Vents	Relocate	Relocate	

TABLE 1 (Continued)

WATTS BAR UNIT 1 SUMMARY

Emergency Core Cooling Systems

Unit 1 Number	STS-Rev 5 Number	Watts Bar Unit 1 Title (STS Rev. 5 for WB N/A)	NRC Results	Unit 1 Results	Notes
3.5.1.1	3.5.1.1	Accumulators - Cold Leg Injection	Retain	Retain	
3.5.1.2	3.5.1.2	Upper Head Injection	Retain	N/A	6
3.5.2	3.5.2	ECCS Subsystem - T_{avg} $\geq$ 350°F	Retain	Retain	
3.5.3	3.5.3	ECCS Subsystem - T_{avg} < 350°F	Retain	Retain	
N/A	3.5.4.1	Boron Injection Tank	Retain	N/A	6
N/A	3.5.4.2	Heat Tracing	Relocate	N/A	6
3.5.4	3.5.5	Refueling Water Storage Tank	Retain	Retain	

TABLE 1 (Continued)

WATTS BAR UNIT 1 SUMMARY

Containment Systems

Unit 1 Number	STS-Rev 5 Number	Watts Bar Unit 1 Title (STS Rev. 5 for WB N/A)	NRC Results	Unit 1 Results	Notes
3.6.1.1	3.6 1.1	Containment Integrity	Retain	Retain	
3.6.1.2	3.6.1.2	Containment Leakage	Relocate	Relocate	14
3.6.1.3	3.6.1.3	Containment Airlock	Retain	Retain	
N/A	3.6.1.4	Containment Isolation Valve and Channel Weld Pressurization Systems	Relocate	N/A	6
3.6.1.4	3.6.1.5	Internal Pressure	Retain	Retain	
3.6.1.5	3.6.1.6	Air Temperature	Retain	Retain	
3.6.1.6	3.6.1.7	Containment Vessel Structural Integrity	Relocate	Relocate	17
3.6.1.7	3.6.1.8	Shield Building Structural Integrity	Relocate	Retain	18
3.6.1.8	3.6.1.9	Emergency Gas Treatment System	Retain	Retain	
3.6.1.9	3.6.1.10	Containment Ventilation System	Retain	Retain	
3.6.2	3.6.2.1	Containment Spray System	Retain	Retain	
N/A	3.6.2.2	Spray Additive System	Retain	N/A	6
N/A	3.6.2.3	Containment Cooling System	Retain	Relocate	19
N/A	3.6.3	Iodine Cleanup System	Retain	N/A	6
3.6.3	3.6.4	Containment Isolation Valves	Retain	Retain	15
3.6.4.1	3.6.5.1	Hydrogen Monitors	Retain	Retain	
3.6.4.2	3.6.5.2	Electric Hydrogen Recombiners	Retain	Retain	

TABLE 1 (Continued)

WATTS BAR UNIT 1 SUMMARY

Containment Systems (Continued)

Unit 1 Number	STS-Rev 5 Number	Watts Bar Unit 1 Title (STS Rev. 5 for WB N/A)	NRC Results	Unit 1 Results	Notes
3.6.4.3	3.6.5.3	Hydrogen Mitigation System	Retain	Retain	
N/A	3.6.5.4	Hydrogen Purge Cleanup System	Relocate	N/A	6
N/A	3.6.5.5	Hydrogen Mixing System	Retain	N/A	6
N/A	3.6.6	Penetration Room Exhaust Air Cleanup System	Retain	N/A	6
3.6.5.1	3.6.7.1	Ice Bed	Retain	Retain	
3.6.5.2	3.6.7.2	Ice Bed Temperature Monitoring System	Relocate	Relocate	
3.6.5.3	3.6.7.3	Ice Condenser Doors	Retain	Retain	
3.6.5.4	3.6.7.4	Inlet Door Position Monitoring System	Relocate	Relocate	
3.6.5.5	3.6.7.5	Divider Barrier Personnel Access Doors and Equipment Hatches	Retain	Retain	
3.6.5.6	3.6.7.6	Containment Air Return Fan System	Retain	Retain	
3.6.5.7	3.6.7.7	Floor Drains	Retain	Retain	
3.6.5.8	3.6.7.8	Refueling Canal Drains	Retain	Retain	
3.6.5.9	3.6.7.9	Divider Barrier Seal	Retain	Retain	
N/A	3.6.8	Vacuum Relief Valves	Retain	N/A	6

TABLE 1 (Continued)

WATTS BAR UNIT 1 SUMMARY

Plant Systems

Unit 1 Number	STS-Rev 5 Number	Watts Bar Unit 1 Title (STS Rev. 5 for WB N/A)	NRC Results	Unit 1 Results	Notes
3.7.1.1	3.7.1.1	Safety Valves	Retain	Retain	
3.7.1.2	3.7.1.2	Auxiliary Feedwater System	Retain	Retain	
3.7.1.3	3.7.1.3	Condensate Storage Tank	Retain	Retain	
3.7.1.4	3.7.1.4	Specific Activity	Retain	Retain	
3.7.1.5	3.7.1.5	Main Steam Line Isolation Valves	Retain	Retain	
3.7.2	3.7.2	Steam Generator Pressure/ Temperature Limitation	Relocate	Relocate	
3.7.3	3.7.3	Component Cooling Water System	Retain	Retain	
3.7.4	3.7.4	Essential Raw Cooling Water System	Retain	Retain	
3.7.5	3.7.5	Ultimate Heat Sink	Retain	Retain	
3.7.6	3.7.6	Flood Protection Plan	Relocate	Relocate	
3.7.7	3.7.7	Control Room Emergency Ventilation System	Retain	Retain	
3.7.8	3.7.8	Auxiliary Building Gas Treatment System	Retain	Retain	
3.7.9	3.7.9	Snubbers	Relocate	Relocate	
3.7.10	3.7.10	Sealed Source Contamination	Relocate	Relocate	
3.7.11.1	3.7.11.1	Fire Suppression Water System	Relocate	Relocate	7
3.7.11.2	3.7.11.2	Spray and/or Sprinkler Systems	Relocate	Relocate	7

TABLE 1 (Continued)

WATTS BAR UNIT 1 SUMMARY

Plant Systems (Continued)

Unit 1 Number	STS-Rev 5 Number	Watts Bar Unit 1 Title (STS Rev. 5 for WB N/A)	NRC Results	Unit 1 Results	Notes
N/A	3.7.11.4	Halon System	Relocate	N/A	6
3.7.11.4	3.7.11.5	Fire Hose Stations	Relocate	Relocate	7
N/A	3.7.11.6	Yard Fire Hydrant and Hose Houses	Relocate	N/A	6
3.7.12	3.7.12	Fire Rated Assemblies	Relocate	Relocate	7
3.7.13	3.7.13	Area Temperature Monitoring	Relocate	Relocate	

TABLE 1 (Continued)

WATTS BAR UNIT 1 SUMMARY

Electrical Power Systems

Unit 1 Number	STS-Rev 5 Number	Watts Bar Unit 1 Title (STS Rev. 5 for WB N/A)	NRC Results	Unit 1 Results	Notes
3.8.1.1	3.8.1.1	A.C. Sources, Operating	Retain	Retain	
3.8.1.2	3.8.1.2	A.C. Sources, Shutdown	Retain	Retain	
3.8.2.1	3.8.2.1	D. C. Sources, Operating	Retain	Retain	
3.8.2.2	3.8.2.2	D. C. Sources, Shutdown	Retain	Retain	
3.8.3.1	3.8.3.1	Onsite Power Distribution, Operating	Retain	Retain	
3.8.3.2	3.8.3.2	Onsite Power Distribution, Shutdown	Retain	Retain	
3.8.3.3	N/A	Isolation Devices	Not Reviewed	Relocate	
N/A	3.8.4.1	A.C. Circuits Inside Primary Containment	Relocate	N/A	6
3.8.4.1	3.8.4.2	Containment Penetration Conductor Overcurrent Protective Devices	Relocate	Relocate	
3.8.4.2	3.8.4.3	Motor-Operated Valves Thermal Overload Bypass Devices	Relocate	Relocate	
3.8.4.3	N/A	Submerged Component Circuit Protection	Not Reviewed	Relocate	

TABLE 1 (Continued)

WATTS BAR UNIT 1 SUMMARY

Refueling Operations

Unit 1 Number	STS-Rev 5 Number	Watts Bar Unit 1 Title (STS Rev. 5 for WB N/A)	NRC Results	Unit 1 Results	Notes
3.9.1	3.9.1	Boron Concentration	Retain	Retain	
3.9.2	3.9.2	Instrumentation	Retain	Retain	
3.9.3	3.9.3	Decay Time	Retain	Relocate	16
3.9.4	3.9.4	Containment Building Penetrations	Retain	Retain	
3.9.5	3.9.5	Communications	Relocate	Relocate	
3.9.6	3.9.6	Refueling Machine	Relocate	Relocate	
3.9.7	3.9.7	Crane Travel - Spent Fuel Storage Pool Building	Relocate	Relocate	
3.9.8.1	3.9.8.1	Residual Heat Removal and Coolant Circulation, High Water Level	Retain	Retain	
3.9.8.2	3.9.8.2	Refueling Operations, Low Water Level	Retain	Retain	
3.9.9	3.9.9	Containment Ventilation Isolation System	Retain	Retain	
3.9.10	3.9.10	Water Level - Reactor Vessel	Retain	Retain	
3.9.11	3.9.11	Water Level - Storage Pool	Retain	Retain	
3.9.12	3.9.12	Auxiliary Building Gas Treatment System	Retain	Retain	
3.9.13	N/A	Reactor Building Purge Ventilation System	Not Reviewed	Retain	

TABLE 1 (Continued)

WATTS BAR UNIT 1 SUMMARY

Special Test Exceptions

Unit 1 Number	STS-Rev 5 Number	Watts Bar Unit 1 Title (STS Rev. 5 for WB N/A)	NRC Results	Unit 1 Results	Notes
3.10.1	3.10.1	Shutdown Margin	Relocate	Retain	1
3.10.2	3.10.2	Group Height, Insertion, and Power Distribution Limits	Retain	Retain	
3.10.3	3.10.3	Physics Tests	Retain	Retain	
3.10.4	3.10.4	Reactor Coolant Loops	Retain	Retain	
N/A	3.10.5	Position Indication System - Shutdown	Relocate	N/A	6

TABLE 1 (Continued)

WATTS BAR UNIT 1 SUMMARY

Radioactive Effluents

Unit 1 Number	STS-Rev 5 Number	Watts Bar Unit 1 Title (STS Rev. 5 for WB N/A)	NRC Results	Unit 1 Results	Notes
3.11.1.1	3.11.1.1	Concentration	Relocate	Relocate	8
3.11.1.2	3.11.1.2	Dose	Relocate	Relocate	8
3.11.1.3	3.11.1.3	Liquid Radwaste Treatment System	Relocate	Relocate	8
3.11.1.4	3.11.1.4	Liquid Holdup Tanks	Relocate	Relocate	8
3.11.2.1	3.11.2.1	Gaseous Effluents, Dose Rate	Relocate	Relocate	8
3.11.2.2	3.11.2.2	Dose - Noble Gases	Relocate	Relocate	8
3.11.2.3	3.11.2.3	Dose - Iodine 131 and 133, Tritium, and Radioactive Material in Particulate Form	Relocate	Relocate	8
3.11.2.4	3.11.2.4	Gaseous Radwaste Treatment System	Relocate	Relocate	8
3.11.2.5	3.11.2.6	Explosive Gas Mixture	Relocate	Relocate	8
3.11.2.6	3.11.2.5	Gas Decay Tanks	Relocate	Relocate	8
3.11.3	3.11.3	Solid Radioactive Wastes	Relocate	Relocate	8
3.11.4	3.11.4	Total Dose	Relocate	Relocate	8

TABLE 1 (Continued)

WATTS BAR UNIT 1 SUMMARY

Radiological Environmental Monitoring

Unit 1 Number	STS-Rev 5 Number	Watts Bar Unit 1 Title (STS Rev. 5 for WB N/A)	NRC Results	Unit 1 Results	Notes
3.12.1	3.12.1	Monitoring Program	Relocate	Relocate	8
3.12.2	3.12.2	Land Use Census	Relocate	Relocate	8
3.12.3	3.12.3	Interlaboratory Comparison Program	Relocate	Relocate	8

NOTES

1) Although NRC letter, Reference 2 allows relocation of LCO 3.1.1.1, Shutdown Margin, Tavg > 200°F for Modes 1 and 2, the WOG has retained this requirement in Technical Specifications. Additionally, Special Test Exception 3.10.1, Shutdown Margin, which applies to the Mode 2 requirements of LCO 3.1.1.1 must also be retained in Technical Specifications with the LCO requirement.

2) Reactor physics parameter LCO limits will be relocated to the Core Operating Limits Report (COLR) in accordance with the guidance provided in NRC Generic Letter 88-16, Reference 4. The COLR will be controlled by requirements specified in the Administrative Controls section of Technical Specifications.

3) NRC letter, Reference 2 allows relocation of LCO 3.1.3.2, Position Indicating Systems - Operating, but specifies retention of the associated surveillance requirements. The entire specification will be retained.

4) LCO 3.1.3.3, Position Indicating Systems - Shutdown, and LCO 3.1.3.4, Rod Drop Time, surveillance requirements will be retained in the Technical Specifications in accordance with NRC letter, Reference 2. LCO 3.1.3.4 will not be relocated since it serves no purpose, i.e., the retained surveillance and its new LCO fulfill the necessary required actions. LCO 3.1.3.3 will be relocated using the version contained in the 1985 Draft T.S. for Analog systems rather than the STS version for digital RPIs.

5) The radiation effluent monitoring instrumentation from LCO 3.3.3.1, Radiation Monitoring Instrumentation, will be relocated to the Administrative Controls section of Technical Specifications (ODCM) consistent with the guidance provided in NRC letter, Reference 2, and NRC Generic Letter 89-01, Reference 3. The remaining instruments will be retained in Technical Specifications.

6) This specification is in STS Rev. 5, but is not applicable to Watts Bar.

7) Fire protection LCOs have been relocated to the TRM.

8) This LCO will be relocated to the Administrative Controls section of Technical Specifications (ODCM, PCP, or other programs) in accordance with NRC letter, Reference 2, and the provisions of NRC Generic Letter 89-01, Reference 3.

   The Liquid Holdup Tanks and Explosive Gas Mixture LCO requirements will be relocated to the Administrative Controls section of Technical Specifications as a new program, Explosive Gas and Storage Tank Radioactivity Monitoring Program, in accordance with meeting agreements between the NRC and NUMARC/WOG.

NOTES (Continued)

9) The LCO 3.3.3.9, Radioactive Gaseous Effluent Monitoring Instrumentation, radioactive gaseous effluent monitoring instrumentation will be relocated to the Administrative Controls section of Technical Specifications (ODCM) in accordance with NRC letter, Reference 2, and the provisions of NRC Generic Letter 89-01, Reference 3. The explosive gas monitoring instrumentation will be relocated to the Administrative Controls section of Technical Specifications as a new program, Explosive Gas and Storage Tank Radioactivity Monitoring Program, in accordance with meeting agreements between the NRC and NUMARC/WOG.

10) The WBN 1985 draft Technical Specification LCO 3.3.4, Turbine Overspeed Protection, is comparable to the STS - Rev. 5 LCO 3.3.4 except that the STS surveillance 4.3.4.2 is not included. The surveillance requirements are specified in the FSAR. The NRC result that this LCO may be relocated still applies without further evaluation.

11) LCO 3.4.5, Steam Generators, and the associated surveillance requirements for tube inspection will be relocated to the Administrative Controls section of Technical Specifications as a program in accordance with NRC letter, Reference 2. The ASME Section XI inservice inspection surveillance requirements will also be covered by new Administrative Controls program.

12) LCO 3.4.9.1, Pressure/Temperature Limits, Reactor Coolant System, LCO limits will be relocated to the Pressure/Temperature Limits Report (PTLR) and controlled by requirements specified in the Administrative Controls section of Technical Specifications. The PTLR and the Administrative Controls will be consistent with the guidance provided by NRC Generic Letter 88-16, Reference 4, for removal of cycle-specific parameter limits (reactor physics parameters) from Technical Specifications.

13) LCO 3.4.10, Structural Integrity, surveillance requirements for Reactor Coolant Pump flywheel inspection will be relocated to the Administrative Controls section of Technical Specifications in accordance with NRC letter, Reference 2. The ASME Section XI inservice inspection surveillance requirements will be covered by a new Administrative Controls program, which will also include the flywheel inspection requirements.

14) LCO 3.6.1.2, Containment Leakage, and the associated surveillance requirements will be relocated to the Administrative Controls section of Technical Specifications as a program; however, the leak rate test parameters including Pa and La will be retained in the Containment Technical Specification Bases in accordance with NRC letter, Reference 2.

15) LCO 3.6.3, Containment Isolation Valves, and the associated surveillance requirements will be retained except that the response times of containment isolation valves will be relocated in accordance with NRC letter, Reference 2.

16) Although application of the selection criteria to LCO 3.9.3, Decay Time, showed that the LCO should be retained, it was agreed in meetings between NUMARC/WOG and the NRC that this LCO may be relocated and it is not contained in NUREG-1431. Therefore, LCO 3.9.3, Decay Time will be relocated to the TRM.

17) The LCO was deleted and the surveillance requirement was retained with the LCO for Containment Operability in NUREG-1431. The new LCO provides the appropriate actions, therefore, the old LCO was deleted as unnecessary.

18) NRC Letter, Reference 2 allows relocation of LCO 3.6.1.7, Shield Building Structural Integrity, but specifies retention of the associated surveillance requirements. The entire specification will be retained and is also contained in NUREG-1431.

19) The Containment Cooling system, described in STS Rev. 5 and evaluated in References 1 and 2, is redundant to the containment spray system which is required for LOCA events. The Watts Bar Lower Compartment Cooling (LCC) system does not perform a heat removal function for DBAs. The LCC system may be used several hours after a non-LOCA event to provide air circulation to eliminate "hot spots" which could affect equipment qualification, as discussed in FSAR Section 6.2.2. The 1985 Draft TS did not contain an LCO for the LCC system, however, one was added to the Sequoyah Nuclear Plant Technical Specifications after this problem was discovered. The "relocated LCO" is based on LCO 3.6.2.2 in the Sequoyah T.S.

TABLE 2

CRITERIA APPLICATION SCREENING FORM

LCO 3.6.2.3
(STS Rev. 5)

I. TECHNICAL SPECIFICATION: Lower Compartment Cooling System

II. EVALUATION BASED ON NRC POLICY STATEMENT CRITERIA

A. Is the Structure, System, or Parameter applicable to:

		YES	NO
1.	Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.	____	X
2.	A process variable that is an initial condition of a Design Basis Accident (DBA) or Transient Analyses that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.	____	X
3.	A structure, system or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.	____	X

B. If the answer to any one of the above questions is "YES" then the Structure, System, or Parameter shall be included in the Technical Specifications.

C. If the answer to all three of the above questions is "NO" and the Structure, System, or Parameter has not been shown to be a significant risk contributor, then it is not required to be included in the Technical Specifications.

	N/A	YES	NO
The Structure, System, or Parameter has been shown to be a significant risk contributor.	____	____	X

TABLE 2 (Continued)

CRITERIA APPLICATION SCREENING FORM

LCO 3.6.2.3
(STS Rev. 5)

III. DISCUSSION

The Lower Compartment Cooling (LCC) System provides air mixing of lower containment dead-end compartments to prevent localized "hot spots" caused by a non-LOCA event. This mixing ensures that containment temperatures remain below environmental qualification limits for all safety related equipment required after a non-LOCA event. Other systems available during a LOCA or non-LOCA event are the Ice Condenser, Air Return Fan, and Containment Spray and RHR Spray. The Ice Condenser and Air Return Fan systems are considered the primary means of cooling Containment following a LOCA or non-LOCA event. During normal operations the Lower Compartment Coolers mix and cool the air in the lower containment compartments.

The LCC fans are not installed instrumentation that is used to detect, and indicate in the control room , a significant abnormal degradation of the reactor coolant pressure boundary. The LCC fans do not satisfy Criterion 1.

The LCC fans are not process variables that are an initial condition of a DBA or Transient Analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. The LCC fans do not satisfy Criterion 2.

The post-accident function of the LCC fans is to support the equipment which operates in the lower containment compartments. For Watts Bar, the LCC system is not required to provide diverse heat removal capabilities beyond the Ice Condenser, Containment Spray, and Air Return Fan systems. Therefore, the LCC fans are not a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a DBA or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. The LCC fans do not satisfy Criterion 3.

Based on the Watts Bar Unit 1 PRA Report in Appendix A, the Lower Compartment Cooling System Fans have not been identified as a significant risk contributor.

IV. CONCLUSION

_____ The Structure, System, or Parameter shall be included in the Technical Specifications.

__X__ The Structure, System, or Parameter is not required to be included in the Technical Specifications.

TABLE 2 (Continued)

CRITERIA APPLICATION SCREENING FORM

LCO 3.8.3.3

I. TECHNICAL SPECIFICATION: Isolation Devices

STS Location: None

II. EVALUATION BASED ON NRC POLICY STATEMENT CRITERIA

A. Is the Technical Specification applicable to:

		YES	NO
1.	Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.		X
2.	A process variable that is an initial condition of a Design Basis Accident (DBA) or Transient Analyses that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.		X
3.	A structure, system or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.		X

B. If the answer to any one of the above questions is "YES" then the Structure, System, or Parameter shall be included in the Technical Specifications.

C. If the answer to all three of the above questions is "NO" and the Structure, System, or Parameter has not been shown to be a significant risk contributor, then it is not required to be included in the Technical Specifications.

	N/A	YES	NO
The Structure, System, or Parameter has been shown to be a significant risk contributor.			X

TABLE 2 (Continued)

CRITERIA APPLICATION SCREENING FORM

LCO 3.8.3.3

III. DISCUSSION

The function of isolation devices is to protect the 1E busses from current faults generated by nonqualified loads powered by the 1E busses. The isolation devices are circuit breakers which open upon receipt of a fault current generated by nonqualified loads on the 1E busses.

The isolation devices are not installed instrumentation that is used to detect, and indicate in the control room , a significant abnormal degradation of the reactor coolant pressure boundary. The isolation devices do not satisfy Criterion 1.

The isolation devices are not process variables that are an initial condition of a DBA or Transient Analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. The isolation devices do not satisfy Criterion 2.

The isolation devices are not a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a DBA or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. The isolation devices do not satisfy Criterion 3.

Based on the Watts Bar Unit 1 PRA Report contained in Appendix A, the Isolation Devices have not been identified as a significant risk contributor.

IV. CONCLUSION

_____ The Structure, System, or Parameter shall be included in the Technical Specifications.

__X__ The Structure, System, or Parameter is not required to be included in the Technical Specifications.

TABLE 2 (Continued)

CRITERIA APPLICATION SCREENING FORM

LCO 3.8.4.3

I. TECHNICAL SPECIFICATION: Submerged Component Circuit Protection

STS Location: None

II. EVALUATION BASED ON NRC POLICY STATEMENT CRITERIA

A. Is the Structure, System, or Parameter applicable to:

		YES	NO
1.	Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.		X
2.	A process variable that is an initial condition of a Design Basis Accident (DBA) or Transient Analyses that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.		X
3.	A structure, system or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.		X

B. If the answer to any one of the above questions is "YES" then the Structure, System, or Parameter shall be included in the Technical Specifications.

C. If the answer to all three of the above questions is "NO" and the Structure, System, or Parameter has not been shown to be a significant risk contributor, then it is not required to be included in the Technical Specifications.

	N/A	YES	NO
The Structure, System, or Parameter has been shown to be a significant risk contributor.			X

TABLE 2 (Continued)

CRITERIA APPLICATION SCREENING FORM

LCO 3.8.4.3

III. DISCUSSION

The submerged component circuit protection de-energizes those valves and motors which are below the post LOCA flood line. De-energizing the valves and motors serves two functions:

1. The de-energized component will not adversely affect the operation of the system it is a part of.

2. Class 1E distribution systems cannot be adversely affected by electrical faults that may be caused by submerged valves or motors.

The Submerged Component Circuit Protection is not installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary. The Submerged Component Circuit Protection does not satisfy Criterion 1.

The Submerged Component Circuit Protection is not a process variable that is an initial condition of a DBA or Transient Analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. The Submerged Component Circuit Protection does not satisfy Criterion 2.

The Submerged Component Circuit Protection is not a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a DBA or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. The Submerged Component Circuit Protection does not satisfy Criterion 3.

Based on Appendix A, the Submerged Component Circuit Protection has not been identified as a significant risk contributor.

IV. CONCLUSION

_____ The Structure, System, or Parameter shall be included in the Technical Specifications.

__X__ The Structure, System, or Parameter is <u>not</u> required to be included in the Technical Specifications.

TABLE 2 (Continued)

CRITERIA APPLICATION SCREENING FORM

LCO 3.9.13

I. TECHNICAL SPECIFICATION: Reactor Building Purge Ventilation System

STS Location: None

II. EVALUATION BASED ON NRC POLICY STATEMENT CRITERIA

A. Is the Structure, System, or Parameter applicable to:

		YES	NO
1.	Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.		X
2.	A process variable that is an initial condition of a Design Basis Accident (DBA) or Transient Analyses that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.		X
3.	A structure, system or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.	X	

B. If the answer to any one of the above questions is "YES" then the Structure, System, or Parameter shall be included in the Technical Specifications.

C. If the answer to all three of the above questions is "NO" and the Structure, System, or Parameter has not been shown to be a significant risk contributor, then it is not required to be included in the Technical Specifications.

	N/A	YES	NO
The Structure, System, or Parameter has been shown to be a significant risk contributor.	X		

TABLE 2 (Continued)

CRITERIA APPLICATION SCREENING FORM

LCO 3.9.13

III. DISCUSSION

Limitations on the operation of the Reactor Building Purge Ventilation System ensure that all radioactive material released inside containment is filtered prior to discharge to the atmosphere.

Although the purge function of the Reactor Building Purge Ventilation System is not a safety-related function, the system does provide a safety-related filtration path. Radioactive gases released from ruptured fuel rods shall be filtered through HEPA filters and charcoal absorbers. This filtration path mitigates the results of a fuel handling accident inside containment, until the associated containment isolation valves are closed.

The Reactor Building Purge Ventilation System is not installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary. The Reactor Building Purge Ventilation System does not satisfy Criterion 1.

The Reactor Building Purge Ventilation System is not process variable that is an initial condition of a DBA or Transient Analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. The Reactor Building Purge Ventilation System does not satisfy Criterion 2.

The Reactor Building Purge Ventilation System is a structure, system or component that is part of the primary success path and which functions or actuates to mitigate a DBA or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. The Reactor Building Purge Ventilation System satisfies Criterion 3.

IV. CONCLUSION

__X__ The Structure, System, or Parameter shall be included in the Technical Specifications.

_____ The Structure, System, or Parameter is <u>not</u> required to be included in the Technical Specifications.

TABLE 3

DISPOSITION OF RELOCATED SPECIFICATIONS OR REQUIREMENTS

LCOs Relocated to the TRM

The following LCOs will be relocated to the TRM.

Unit 1 Number	Title	Notes
3.1.2.1	Flow Paths, Shutdown	
3.1.2.2	Flow Paths, Operating	
3.1.2.3	Charging Pump, Shutdown	
3.1.2.4	Charging Pump, Operating	
3.1.2.5	Borated Water Sources - Shutdown	
3.1.2.6	Borated Water Sources - Operating	
3.3.3.2	Movable Incore Detectors	
3.3.3.3	Seismic Instrumentation	
3.3.3.7	Fire Detection Instrumentation	
3.3.3.10	Loose Part Detection System	
3.3.4	Turbine Overspeed Protection	
3.4.2.1	Safety Valves - Shutdown	
3.4.7	Chemistry	
3.4.9.2	Pressure/Temperature Limits, Pressurizer	
3.4.11	Reactor Coolant System Vents	
N/A	Lower Compartment Cooling System Fans	Not contained in 1985 Watts Bar Tech Specs
3.6.5.2	Ice Bed Temperature Monitoring System	
3.6.5.4	Inlet Door Position Monitoring System	

Unit 1 Number	Title	Notes

TABLE 3 (Continued)

DISPOSITION OF RELOCATED SPECIFICATIONS OR REQUIREMENTS

LCOs Relocated to the TRM (Continued)

Unit 1 Number	Title	Notes
3.7.1	Flood Protection Plan	
3.7.2	Steam Generator Pressure/Temperature Limitation	
3.7.9	Snubbers	
3.7.10	Sealed Source Contamination	
3.7.11.1	Fire Suppression Water System	
3.7.11.2	Spray and/or Sprinkler System	
3.7.11.3	CO_2 Systems	
3.7.11.4	Fire Hose Stations	
3.7.12	Fire Rated Assemblies	
3.7.13	Area Temperature Monitoring	
3.8.3.3	Isolation Devices	
3.8.4.1	Containment Penetration Conductor Overcurrent Protective Devices	
3.8.4.2	Motor-Operated Valves Thermal Overload Bypass Devices	
3.8.4.3	Submerged Component Circuit Protection	
3.9.3	Decay Time	
3.9.5	Communications	
3.9.6	Refueling Machines	
3.9.7	Crane Travel - Spent Fuel Storage Pool Building	

TABLE 3

DISPOSITION OF RELOCATED SPECIFICATIONS OR REQUIREMENTS

Relocated LCOs - Retained Surveillance Requirements

The following LCOs will be relocated to the TRM, however, the surveillance requirements will be retained in a related specification.

Unit 1 Number	Title	Notes
3.1.3.3	Position Indicating Systems - Shutdown	The LCO for analog RPIs used in the 1985 Draft T.W. was relocated, not the digital version in STS Rev. 5.
3.1.3.4	Rod Drop Time	The LCO was deleted and the surveillance requirement was retained.
3.6.1.6	Containment Vessel Structural Integrity	The LCO was deleted and the surveillance requirement was retained.

TABLE 3 (Continued)

DISPOSITION OF RELOCATED SPECIFICATIONS OR REQUIREMENTS

RETS LCOs Relocated to Programs

The following LCOs will be relocated to the Administrative Controls section of the Technical Specifications (ODCM, PCP or Explosive Gas and Storage Tank Radioactivity Monitoring Program) in accordance with NRC letter, Reference 2, the provisions of NRC Generic Letter 89-01, Reference 3 and agreements from meetings between NUMARC/WOG and the NRC.

Unit 1 Number	Title	Notes
3.3.3.4	Meteorological Instrumentation	
3.3.3.8	Radioactive Liquid Effluent Monitoring Instrumentation	
3.3.3.9	Radioactive Gaseous Effluent Monitoring Instrumentation	Radioactive gaseous effluent monitoring instrumentation only
3.11.1.1	Concentration	
3.11.1.2	Dose	
3.11.1.3	Liquid Radwaste Treatment System	
3.11.1.4	Liquid Holdup Tanks	
3.11.2.1	Gaseous Effluents, Dose Rate	
3.11.2.2	Dose - Noble Gases	
3.11.2.3	Dose - Iodine 131 and 133, Tritium, and Radioactive Material in Particulate Form	
3.11.2.4	Gaseous Radwaste Treatment System	
3.11.2.5	Explosive Gas Mixture	
3.11.2.6	Gas Decay Tanks	
3.11.3	Solid Radioactive Waste	
3.11.4	Total Dose	
3.12.1	Monitoring Program	
3.12.2	Land Use Census	
3.12.3	Interlaboratory Comparison Program	

TABLE 3 (Continued)

DISPOSITION OF RELOCATED SPECIFICATIONS OR REQUIREMENTS

LCOs Relocated to Programs

The following LCOs or requirements will be specified in the Design Features or Administrative Controls sections of Technical Specifications as indicated in the Notes Column below.

Unit 1 Number	Title	Notes
3.4.5	Steam Generators	Tube inspection will be relocated to the Administrative Controls section as a program. The ASME Section XI surveillance requirements will be relocated to the Administrative Controls section as part of the Inservice Inspection Program.
3.4.10	Structural Integrity	RCS pump flywheel inspection and ASME Section XI surveillance requirements will be relocated to the Administrative Controls Inservice Inspection Program.
3.6.1.2	Containment Leakage	This LCO will be relocated to the Administrative Controls section as a program. The leak test parameters will be retained in the Containment Bases.

TABLE 3 (Continued)

DISPOSITION OF RELOCATED SPECIFICATIONS OR REQUIREMENTS

LCO Limits Relocated to COLR

Reactor physics parameter LCO limits from the following specifications will be relocated to the Core Operating Limits Report (COLR) in accordance with the guidance provided in NRC Generic Letter 88-16, Reference 4. The COLR will be controlled by requirements specified in the Administrative Controls section of Technical Specifications.

Unit 1 Number	Title	Notes
3.1.1.3	Moderator Temperature Coefficient	
3.1.3.1	Movable Control Assemblies, Group Height	
3.1.3.5	Shutdown Rod Insertion Limit	
3.1.3.6	Control Rod Insertion Limits	
3.2.1	Axial Flux Difference	
3.2.2	Heat Flux Hot Channel Factor - $F_Q(Z)$	
3.2.3	RCS Flow Rate and Nuclear Enthalpy Rise Hot Channel Factor	

TABLE 3 (Continued)

DISPOSITION OF RELOCATED SPECIFICATIONS OR REQUIREMENTS

LCO Limits Relocated to the RCS PTLR

Pressure/Temperature LCO limits from the following specifications will be relocated to the RCS Pressure/Temperature Limits Report (RCS PTLR) consistent with the guidance provided in NRC Generic Letter 88-16, Reference 4, for physics parameters. The PTLR will be controlled by requirements specified in the Administrative Controls section of Technical Specifications.

Unit 1 Number	Title	Notes
3.4.9.1	Pressure/Temperature Limits, Reactor Coolant System	The RCS temperature rates of change limits have been relocated to the PTLR. The RCS P/T limits for heatup, cooldown, inservice hydrostatic and leak testing, and criticality have been relocated to the PTLR.
3.4.9.3	Overpressure Protection Systems	The PORV lift settings have been relocated to the PTLR.

APPENDIX A

PROBABILISTIC RISK ASSESSMENT (PRA)

EVALUATION OF THE WATTS BAR UNIT 1 TECHNICAL SPECIFICATIONS

I. OBJECTIVE

This Appendix documents the Probabilistic Risk Assessment (PRA) evaluation of the Watts Bar Unit 1 Technical Specifications in order to support implementation of the MERITS Program. Per the Interim Policy Statement, Technical Specifications must be evaluated from a PRA point of view. Thus, the purpose of this analysis is to determine if the parameters, components, or systems addressed by the Watts Bar Unit 1 Technical Specifications, have been modeled within the available literature on risk insights and PRA studies, and whether they are of prime importance in limiting the likelihood or severity of accident sequences commonly found to dominate offsite health effects.

II. EVALUATION BASES

Three criteria in addition to the PRA evaluation process are included in the Interim Policy Statement for determining which specifications are to be retained in the Technical Specifications. These three criteria plus the PRA evaluation were applied to the Standard Technical Specifications (STS) for Westinghouse Pressurized Water Reactors (NUREG 0452, Revision 4 and draft Revision 5) and documented in Reference 1. If none of the three criteria was identified as a constraint for a given specification, then that specification was identified as a candidate for relocation to another controlled document. Specifications identified for possible relocation were then evaluated using PRA experience to determine if the parameters, components, or systems addressed by the specifications have been modeled within the available literature on risk insights and PRA studies, and whether they are of prime importance in limiting the likelihood or severity of accident sequences commonly found to dominate offsite health effects. If these specifications were not found to be of prime importance in limiting the likelihood or severity of accident sequences that dominate offsite health effects, then these specifications could be relocated to another controlled document.

The results of the MERITS criteria application to the STS were submitted to the NRC by the Westinghouse Owners Group. The NRC then issued a position statement on the results identifying which specifications in the STS must be retained or may be relocated. These NRC determinations (Ref. 2) formed the basis for the application of the Interim Policy Statement criteria and the PRA evaluation to the Watts Bar Unit 1 Technical Specifications.

In the main section of this report, the Watts Bar Unit 1 Technical Specifications were compared to the STS. For those specifications with comparable requirements, the disposition followed the NRC position statement. For those requirements unique to Watts Bar Unit 1 or inconsistent with the STS, the Interim Policy Statement criteria were applied. Table A-1 identifies the Watts Bar Unit 1 plant specific Technical Specification relocation candidates for which a PRA study is required. As directed by the NRC Interim Policy Statement, these must also be evaluated using PRA experience to determine if the parameters, components, or systems addressed by the specifications have been modeled within the available literature on risk insights and PRA studies, and whether they are of prime importance in limiting the likelihood or severity of accident sequences commonly found to dominate offsite health effects. If these specifications are not found to be of prime importance in limiting the likelihood or severity of events that dominate

offsite health effects, then they may be relocated to another controlled document. In this way, a decision to retain or relocate each of the Technical Specifications for Watts Bar Unit 1 was obtained.

The evaluation of the risk impact of the Watts Bar Unit 1 Technical Specifications that are relocation candidates (Table A-1), is based on the following:

A. It is assumed that any Technical Specification that is relocated will be transferred to other utility controlled documents.

B. The risk criteria used in determining the disposition of a Technical Specification are the following:

   1. If the Technical Specification contains constraints of <u>prime importance</u> in limiting the likelihood or severity of the accident sequences <u>commonly found to dominate</u> offsite health effects, it should be retained.

   2. If the Technical Specification covers items modeled in a dominant sequence but has an insignificant impact on the probability or severity of that sequence, it may be relocated to another controlled document.

   3. If the Technical Specification is not modeled in risk dominant sequences, it may be relocated to another controlled document.

C. The measures related to risk used in this evaluation are core melt frequency and offsite health effects. These measures are consistent with the Interim Policy Statement on Technical Specifications and the Safety Goal and Severe Accident Policy Statements.

D. The numerical risk evaluation criteria used to determine if a sequence is risk dominant are:

   1. For core melt, any sequence whose frequency is found to be greater than 2.7×10^{-7} per reactor year is maintained as a possible dominant sequence as a conservative first cut. This is roughly 1% of the total core melt frequency of 2.7×10^{-5} for a plant like Watts Bar (Ref. 32). Each specific sequence found to be greater than 2.7×10^{-7} per reactor year is then evaluated based on the offsite health effects.

   2. For offsite health effects, the Sequoyah Nuclear Plant Individual Plant Evaluation (Ref 32) was used. This evaluation shows that the accident sequences which dominate core damage typically do not coincidently result in a failure of the containment system. Only four sequences (sequences 3, 9, 10, & 12 in Table A-2) could lead to containment failure and result in a significant fission product release.

E. Table A-2 lists sequences for a Westinghouse four-loop, ice condenser containment plant like Watts Bar Unit 1, (Ref 32). The table has been

reviewed for consistency with NRC sponsored PRA programs documented in NUREG's 3301, 1150, 4550, 4551, 4624, and 4700 (Ref. 7-12) and the results have been found to be consistent. For example, these NRC sponsored PRA programs have identified similar accident sequences, such as: transients with loss of support systems leading to reactor coolant pump seal LOCAs, loss of offsite power with failure to recover power, LOCAs with failure of injection or recirculation, and loss of support systems (vital AC/DC, service water and component cooling water). Systems identified in Table A-2 that contribute significantly to risk as defined in item D above are listed in Tables A-2a and A-2b. These identified systems were used to screen the requirements of the Technical Specifications identified in Table A-1 as candidates for relocation. Those specifications whose requirements were relevant to these systems, sequences, and initiating events were further evaluated for risk dominance. The remaining specifications were evaluated on the basis of risk insights from references that were not formal full scale risk assessments. If the requirements of a Technical Specification were not found to be modeled in any reference and no significant issues were identified from a review of the risk insights, the conclusion was that it did not contain constraints of prime importance to limiting the likelihood or severity of sequences commonly found to dominate offsite health effects.

III. METHODOLOGY

The formal process used in evaluating the risk significance of the Technical Specifications is outlined below:

A. The requirements of the Technical Specifications were screened against the dominant sequences and initiating events of Tables A-2, A-2a and A-2b to identify if they were pertinent to risk dominant sequences. The conclusion of this screening was stated at the beginning of the "COMMENTS" section of the review form discussed below.

B. The references used for the STS PRA evaluation (Ref. 1) were reviewed based on the Watts Bar Unit 1 design and the specifications requiring a PRA evaluation identified in Table A-1. Those references identified as providing possible PRA insights applicable to the Watts Bar Unit 1 evaluation were selected to be used in the PRA evaluation process. A list of all reference documents used in this evaluation is provided in Section V.

C. PRA review forms (page A-21) were developed which formalized the PRA review. These review forms contain:

1. The number and title of the Technical Specification;

2. A description of the Technical Specification requirement;

3. The potential safety effect of the Technical Specification requirement;

4. The reference documents that were utilized to support the comments and conclusions on the Technical Specification requirement;

5. Comments, which contain a discussion of the information that was used as the basis to arrive at the conclusion;

6. A conclusion as to whether the Technical Specification should be retained or relocated.

D. The Watts Bar Unit 1 specifications listed in Table A-1 were screened to determine which reference documents addressed the requirements of that specification. All of the documents listed in Section V were considered as part of this review. The screening of references was performed as discussed in Section II under item E.

If there were no reference documents found as a basis for evaluating a specification it was indicated on the review form by entering "None" under the PRA Design Basis Reference column. If constraints of a specification were not modeled in any formal risk assessments, the references which were reviewed to confirm this were listed.

E. A PRA review form was completed for each Watts Bar Unit 1 Technical Specification that did not meet at least one of the Interim Policy Statement screening criteria (Table A-1). These review forms, containing the information and the reference documentation that were used to arrive at the PRA conclusion are documented in this Appendix.

F. As a result of the PRA screening, a conclusion was reached as to whether the Technical Specification:

1. Should be retained as a Technical Specification;
2. May be relocated to other utility controlled documents.

This methodology is based on the PRA methodology presented in the MERITS Program Criteria Application report (Ref. 1). However, several changes were made to the original PRA methodology. The PRA results used to identify dominant sequences and systems for a subatmospheric containment were replaced with PRA results from a Westinghouse 4-loop plant with an ice condenser containments to be more representative of Watts Bar Unit 1. No senior review group was established since the specifications requiring review were significantly less in number and fairly similar to the Standard Technical Specifications. Only formal PRA assessments were used as references.

IV. PRA EVALUATION RESULTS

The results of the PRA evaluation are shown in Table A-3. A "Retain" entry in the PRA evaluation column indicates the Technical Specification requirements contain restraints of prime importance in limiting the likelihood or severity of accident sequences commonly found to dominate offsite health effects. A "Relocate" entry in the PRA evaluation column indicates that the Technical Specification requirements do not contain constraints of prime importance in limiting the likelihood or severity of the accident sequences commonly found to dominate offsite health effects.

V. REFERENCES

1. WCAP-11618, "Methodically Engineered, Restructured, and Improved Technical Specifications," MERITS Program - Phase II, Task 5, Criteria Application.

2. NRC Letter, T. Murley to W. Wilgus (B&W Owners Group Chairman), May 9, 1988, (Documenting the NRC Staff review of each Owners Group's Criteria Application Topical Report).

3. 10CFR Part 50, "Policy Statement on Severe Accidents."

4. 10CFR Part 50, "Policy Statement on Safety Goals for the Operation of Nuclear Power Plants."

5. WCAP-10590, PUN Probabilistic Safety Study," 1984.

6. "Millstone Unit 3 Probabilistic Safety Study," Northeast Utilities Company, August 1983.

7. NUREG-1150, "Severe Accident Risks: An Assessment for Five U. S. Nuclear Power Plants," June 1989.

8. NUREG/CR-3301, "Catalog of PRA Dominant Accident Sequence Information," June 1983.

9. NUREG/CR-4550, Volume 3, Rev. 1 "Analysis of Core Damage Frequency Estimation from Internal Events: Surry Unit 1," April 1990.

10. NUREG/CR-4551, Volume 1, Rev. 1, "Evaluation of Severe Accident Risks: Surry Unit 1," October 1990.

11. NUREG/CR-4624, Volume 3, "Radionuclide Release Calculations for Selected Severe Accident Scenarios PWR Subatmospheric Containment Design: Surry Unit 1," July 1986.

12. NUREG/CR-4700, Volume 1, "Containment Event Analysis for Postulated Severe Accidents: Surry Power Station Unit 1," February 1987.

13. NUREG/CR-4550, Volume 5, "Analysis of Core Damage Frequency: Sequoyah, Unit 1," April 1990.

14. NUREG/CR-4550, Volume 7, "Analysis of Core Damage Frequency: Zion Unit 1," May 1990.

15. "Probabilistic Boron Dilution Analysis, South Texas Project," Final Report, September 1986.

16. "Vogtle Electric Generating Plant, Inadvertent Boron Dilution Analysis," Final Report, Rev. 2.

17. "A Probabilistic Safety Analysis of Boron Dilution Events at Millstone Unit 3," Northeast Utilities Service Company.

18. Zion Probabilistic Safety Study," Commonwealth Edison Company, September 1981.

19. NUREG-0460, "Anticipated Transients Without Scram for Light Water Reactors," Volumes 1-4, 1978-80.

20. 10CFR Part 50.62, "Anticipated Transients Without Scram (ATWS)."

21. NUREG-0800, "Standard Review Plan."

22. NRC Regulatory Guide 1.115, "Protection Against Low Trajectory Turbine Missiles," Revision 1, July 1977.

23. NRC Standard Review Plan 3.5.1.3, "Turbine Missiles," Rev. 2, July 1981.

24. "Indian Point Probabilistic Safety Study," Power Authority of the State of New York, Consolidated Edison Company of New York, Inc., 1982.

25. 10CFR Part 50, Appendix A (General Design Criteria 4), "Environmental and Missile Design Bases."

26. WCAP-10161, "Evaluation of Impact of Reduced Testing of Turbine Valves," Westinghouse Electric Corporation, September 1982.

27. WCAP-11525, "Probabilistic Evaluation of Reduction in Turbine Valve Test Frequency," Westinghouse Electric Corporation, June 1987.

28. NUREG/CR-2728, "Interim Reliability Evaluation Program Procedures Guide," 1983.

29. NUREG/CR-2300, "Probabilistic Risk Assessment Procedures Guide," January 1983.

30. NUREG/CR-2815, "Probabilistic Safety Assessment Guide," Rev. 1, August 1985.

31. IDCOR Technical Report 86.3A1, "Individual Plant Evaluation Methodology for Pressurized Water Reactors," April 1987

32. WCAP-11769, "Sequoyah Nuclear Plant Individual Plant Evaluation Report", March 1988

33. NUREG/CR-4551, Vol. 5, Rev. 1, "Evaluation of Severe Accident Risks: Sequoyah, Unit 1," December 1990

TABLE A-1

WATTS BAR UNIT 1 PLANT SPECIFIC TECHNICAL SPECIFICATION
RELOCATION CANDIDATE REQUIRING PRA EVALUATION

Unit 1 LCO	STS-Rev 5 Number	Title	NRC Results	Criteria Results
N/A	None	Lower Compartment Cooling Fans	Not Reviewed	Relocate
3.8.3.3	None	Isolation Devices	Not Reviewed	Relocate
3.8.4.3	None	Submerged Component Circuit Protection	Not Reviewed	Relocate

TABLE A-2

DOMINANT ACCIDENT SEQUENCES FOR A
4-LOOP PLANT WITH AN ICE CONDENSER CONTAINMENT

Rank With Respect To Core Melt	Sequence Description	Mean Annual Core Melt Frequency	Source Term Release Fraction [1]
1	Small LOCA: Failure of High Pressure Recirculation	8.22E-06	<1.0E-03
2	Loss of offsite Power, failure of Auxiliary Feedwater (AFW) and feed and bleed cooling [2]	1.97E-06	<1.0E-03
3	Consequential Small LOCA: Failure of SI, Recirculation Cooling and Cont. Spray	1.85E-06	>1.0E-03
4	Station Blackout with power restored in 7 hours	1.75E-06	<1.0E-03
5	Transient with Failure to Restore Main Feedwater, AFW and Feed & Bleed Cooling	1.44E-06	<1.0E-03
6	Consequential Small LOCA: Failure of High Pressure Recirculation	1.31E-06	<1.0E-03
7	Small LOCA: Failure of High Pressure Recirculation [2]	1.18E-06	<1.0E-03

TABLE A-2 (continued)

DOMINANT ACCIDENT SEQUENCES FOR A
4-LOOP PLANT WITH AN ICE CONDENSER CONTAINMENT

Rank With Respect To Core Melt	Sequence Description	Mean Annual Core Melt Frequency	Source Term Release Fraction [1]
8	Medium LOCA: Failure of High Pressure Recirculation	9.39E-07	<1.0E-03
9	Interfacing System LOCA	8.6E-07	>1.0E-03
10	Consequential Small LOCA: Failure of Safety Inj., Recirc. and Cont. Spray [2]	7.95E-07	>1.0E-03
11	Small LOCA: Loss of High Pressure Recirculation and Containment Spray Recirculation	7.64E-07	<1.0E-03
12	Station Blackout: Failure to Restore Power	6.59E-07	>1.0E-03
13	Station Blackout: Failure of turbine driven AFW pump, power recovered within 3 hours	6.24E-07	<1.0E-03
14	Medium LOCA: Failure of Accumulators	4.73E-07	<1.0E-03

TABLE A-2 (continued)

DOMINANT ACCIDENT SEQUENCES FOR A
4-LOOP PLANT WITH AN ICE CONDENSER CONTAINMENT

Rank With Respect To Core Melt	Sequence Description	Mean Annual Core Melt Frequency	Source Term Release Fraction [1]
15	Transient: Failure of AFW, and Feed and Bleed cooling [2]	4.28E-07	<1.0E-03
16	Transient: Failure of Main Feedwater, AFW and Feed and Bleed Cooling	3.06E-07	<1.0E-03
17	Large LOCA: Failure of Low Pressure Recirculation	2.93E-07	<1.0E-03
18	Small LOCA: Failure of High Pressure Recirculation and Containment Spray Recirculation [2]	2.77E-07	<1.0E-03

Notes

1 Except for Noble Gases

2 Failure of plant support systems (Essential Raw Water Cooling, Component Cooling and AC/DC power) also contributed to the accident sequence failure probability.

General Note

Only those accident sequences with a core melt frequency of $>2.75 \times 10^{-7}$ /yr (>1% of the total core damage) are shown.

TABLE A-2a

PLANT SYSTEM FAILURES CONTRIBUTING TO COREMELT FREQUENCY FOR A 4-LOOP PLANT WITH AN ICE CONDENSER CONTAINMENT CONSIDERING ACCIDENT SEQUENCES WITH FREQUENCY GREATER THAN 1×10^{-7}/YEAR

System
Main Feedwater
Auxiliary Feedwater System
Diesel Generators
Low Pressure Injection
Primary PORVs
Recirculation (High and Low Pressure SI)
Containment Spray Recirculation
Component Cooling Water
Essential Raw Cooling Water
Accumulators

TABLE A-2b

PLANT SYSTEM FAILURES CONTRIBUTING TO PLANT RISK
FOR A 4-LOOP PLANT WITH AN ICE CONDENSER CONTAINMENT
CONSIDERING ACCIDENT SEQUENCES WITH
SOURCE TERM RELEASE FRACTIONS GREATER THAN 1×10^{-3}/YEAR

System
Diesel Generators
Containment Spray and Recirculation
Component Cooling Water
Essential Raw Cooling Water
RHR/RCS Isolation Valves
Vital DC Power

TABLE A-3

PRA EVALUATION OF THE WATTS BAR UNIT 1 PLANT SPECIFIC TECHNICAL SPECIFICATION RELOCATION CANDIDATES

Unit 1 LCO	Title	NRC Results	Criteria Results	PRA Results	Page
N/A	Lower Compartment Cooling Fans	Not Reviewed	Relocate	Relocate	A-21
3.8.3.3	Isolation Devices	Not Reviewed	Relocate	Relocate	A-23
3.8.4.3	Submerged Component Circuit Protection	Not Reviewed	Relocate	Relocate	A-25

PRA TECHNICAL SPECIFICATION REVIEW FORM

TECHNICAL SPECIFICATION:

Lower Compartment Cooling Fans

DESCRIPTION OF REQUIREMENT:

Lower Compartment Cooling (LCC) Fans ensure adequate air circulation capability following non-LOCA accident transients such that the environmental qualification limits of required safety related equipment will not be violated.

POTENTIAL EFFECT:

Lower Containment temperature in excess of the environmental qualification temperature limits.

REFERENCE DOCUMENTS UTILIZED:

PRA DESIGN BASIS	RISK
None	Ref. 13: NUREG-4550, Sequoyah Unit 1, Internal Events Ref. 32: Sequoyah IPE

COMMENTS:

The requirements of this Technical Specification are not risk dominant based on the core melt and health effects screening criteria provided in Section II under item D parts 1 and 2. Further discussion of insights based on review of the reference documents is provided below.

Exceeding environmental qualification temperature limits may adversely impact the operability or accuracy of safety related

equipment. However, the environmental temperature limit is typically well below the failure limit for equipment and the only expected result of the temperature limit violation would be some increase in instrument error to some reduction in the design life of the hardware.

CONCLUSION:

This Technical Specification does not contain constraints of prime importance in limiting the likelihood or severity of the accident sequences commonly found to dominate risk.

RETAIN IN TECHNICAL SPECIFICATIONS	RELOCATE FROM TECHNICAL SPECIFICATIONS
NO	YES
	X

PRA TECHNICAL SPECIFICATION REVIEW FORM

TECHNICAL SPECIFICATION:

3.8.3.3 - Isolation Devices

DESCRIPTION OF REQUIREMENT:

Isolation devices are circuit breakers which actuate upon receipt of fault currents. Fault currents are generated by nonqualified loads that are powered by the 1E busses. The function of the isolation devices are to protect the 1E busses from fault currents.

POTENTIAL EFFECT:

Loss of 1E busses

REFERENCE DOCUMENTS UTILIZED:

PRA DESIGN BASIS	RISK
None	Ref. 6: Millstone Ref. 7: NUREG-1150 Reactor Risk Reference Document Ref. 13 NUREG-4550, Sequoyah Unit 1, Internal Events Ref. 32 Sequoyah IPE

COMMENTS:

The requirements of this Technical Specification are not risk dominant based on the core melt and health effects screening criteria provided in Section II under item D parts 1 and 2. Further discussion of insights based on review of the reference documents is provided below.

Although the Isolation Device failure can lead to the loss of the 1E buss, these specific isolation devices have not been shown to be of prime importance in risk dominant sequences.

CONCLUSION:

This Technical Specification does not contain constraints of prime importance in limiting the likelihood or severity of the accident sequences commonly found to dominate risk.

RETAIN IN TECHNICAL SPECIFICATIONS	RELOCATE FROM TECHNICAL SPECIFICATIONS
NO	YES
	X

PRA TECHNICAL SPECIFICATION REVIEW FORM

TECHNICAL SPECIFICATION:

3.8.4.3 - Submerged Components Circuit Protection

DESCRIPTION OF REQUIREMENT:

The submerged component protection circuits de-energize those valves and motors which will be submerged during a LOCA.

POTENTIAL EFFECT:

REFERENCE DOCUMENTS UTILIZED:

PRA DESIGN BASIS	RISK
None	None

COMMENTS:

The requirements of this Technical Specification are not risk dominant based on the core melt and health effects screening criteria provided in Section II under item D parts 1 and 2.

Submerged component protection circuits are only provided for those components not required to mitigate the effects of LOCA events and thus are not of prime importance in risk dominant sequences.

CONCLUSION:

This Technical Specification does not contain constraints of prime importance in limiting the likelihood or severity of the accident sequences commonly found to dominate risk.

RETAIN IN TECHNICAL SPECIFICATIONS	RELOCATE FROM TECHNICAL SPECIFICATIONS
NO	YES
	X