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WOODY,C.O. Florida Power & Light Co.
RECIP.NAME RECIPIENT AFFILIATION
THOMPSON,H.L. Division of Pressurized Water Reactor Licensing - A (post 8

SUBJECT: Application for amends to License DPr-31 & DPR-41,upgrading
Tech Specs to include guidance of NUREG-0452,STS for
Westinghouse PWRs.Fee paid.

"Pee Tech. Specs."

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FPL

FLORIDA POWER & LIGHT COMPANY

SEPTEMBER 29 1986
L-86-393

Office of Nuclear Reactor Regulation
Attention: Mr. Hugh L. Thompson, Jr., Director
Division of PWR Licensing - A
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Thompson:

Re: Turkey Point Units 3 and 4
Docket Nos. 50-250 and 50-251
Proposed License Amendment
Upgrade of Technical Specifications

In accordance with 10 CFR 50.90, Florida Power and Light Company submits herewith three (3) signed originals and forty copies of a request to amend Appendix A of Facility Operating Licenses DPR 31 and DPR 41.

This amendment is submitted to upgrade the current plant Technical Specifications to include, within limitations, the guidance of NUREG-0452, Standard Technical Specifications for Westinghouse Pressurized Water Reactors. The upgrade of the plant Technical Specifications is the objective of the Turkey Point Plant Performance Enhancement Program Project 10.

The Technical Specifications provided in this revision will result in improved direction to plant operators by providing the Technical Specifications in a format consistent with current standards. In addition, overall plant safety and control of plant systems important to safety will be improved because this upgrade adds many operational and surveillance requirements from the Standard Technical Specifications that are not included in the current plant Technical Specifications. We believe the proposed revisions represent the best program for Turkey Point Units 3 and 4 considering the Technical Specification Revision Project development criteria, i.e. no hardware changes (based on the existing design and analytical basis) or other significant operating hardships.

Attachment I includes the proposed Technical Specification revisions. Not included with this submittal are the technical specifications for Fire Protection and the Electrical Power Systems. Deletion of the existing Fire Protection technical specifications will be done under a separate program consistent with NRC Generic Letter 86-10. Technical Specifications 3/4.8.1 through 3/4.8.3 (Electrical Power Systems) have been developed but are still under review and will be submitted at a later date.

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Attachment II is the Safety Evaluation for the proposed amendment. Appendix B to the Safety Evaluation includes the No Significant Hazards Evaluations that correspond to the individual revised Technical Specification subsections, and provides a basis for the conclusions reached in the Safety Evaluation.

There are significant format changes between the current and revised Technical Specifications. Attachment III is a cross reference between subsections in the current Technical Specifications and the revised Technical Specifications. It is provided to assist you in your review.

The attached revised Technical Specifications represent the plant design configuration as of July 31, 1986. Because the NRC review and approval process for this submittal will be substantial, we anticipate that changes to this submittal may be required prior to final NRC approval. We will work with Mr. D. G. McDonald, the NRC Project Manager for the Turkey Point Plant, to establish a program to incorporate any changes made during the NRC review process.

Current plant operating procedures are being revised to include the additional and more restrictive requirements of the revised Technical Specifications. Any proposed Technical Specification relaxations will be incorporated in plant procedures after NRC review and approval of the Technical Specifications. These procedure revisions, along with operator indoctrination and training, need to be considered in determining an implementation date for the approved revised Technical Specifications. At this time we believe implementation will take approximately 6 months after NRC approval. It should be recognized that the need may arise for emergency or exigent Technical Specification relief in the event that an unanticipated conflict or difficulty emerges during implementation.

In accordance with 10 CFR 50.91(a)(1), it has been determined that the proposed amendment does not involve any significant hazards considerations pursuant to 10 CFR 50.92. The No Significant Hazards Considerations determination is provided in Attachment II.

In accordance with 10 CFR 50.92(b)(1) a copy of the proposed amendment is being forwarded to the State Designee for the State of Florida.

The proposed amendment has been reviewed by the Turkey Point Plant Nuclear Safety Committee and the Florida Power and Light Company Nuclear Review Board.

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In accordance with 10 CFR 170.12(c), FPL Check No. 2344 for the application fee of $150.00 is enclosed.

Very truly yours,

C. O. Woody
Group Vice President
Nuclear Energy

COW/gp

Attachments:

Attachment	I	Revised Technical Specifications
Attachment	II	Safety Evaluation
Attachment	III	Current/Revised Technical Specification Cross Reference

Enclosure

cc: Dr. J. Nelson Grace
Regional Administrator, Region II
U. S. Nuclear Regulatory Commission
101 Marietta Street, N.W., Suite 2900
Atlanta, Georgia 30303

Mr. Alan Schubert
Public Health Physicist
Department of Health and Rehabilitative Services
1323 Winewood Boulevard
Tallahassee, Florida 32301

Harold F. Reis, Esquire

STATE OF FLORIDA)
) ss.
COUNTY OF PALM BEACH)

J. W. Dickey being first duly sworn, deposes and says:

That he is Vice President, Nuclear Operations of Florida Power & Light Company, the Licensee herein;

That he has executed the foregoing document; that the statements made in this document are true and correct to the best of his knowledge, information, and belief, and that he is authorized to execute the document on behalf of said Licensee.

J. W. Dickey

J. W. Dickey

Subscribed and sworn to before me this

29 day of September, 1986.

Bonnie M. Simmons

NOTARY PUBLIC, in and for the County of Palm Beach, State of Florida

NOTARY PUBLIC STATE OF FLORIDA
MY COMMISSION EXP SEPT 18,1989
BONDED THRU GENERAL INS. UND.

My Commission expires: ______________

ATTACHMENT II

TURKEY POINT UNITS 3 AND 4

SAFETY EVALUATION

FOR

REVISED TECHNICAL SPECIFICATIONS

APPENDIX A - TECHNICAL SPECIFICATION IMPROVEMENT MATRIX

APPENDIX B - NO SIGNIFICANT HAZARDS EVALUATIONS

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SAFETY EVALUATION

1.0 BACKGROUND

Turkey Point Units 3 and 4 currently operate with custom technical specifications issued with the operating Licenses in 1972 and 1973. Subsequently the NRC has issued NUREG 0452, Standard Technical Specifications for Westinghouse Pressurized Water Reactors. The Standard Technical Specifications, which have been utilized by new licensed plants are recognized to be more prescriptive and contain an increased number and frequency of surveillances than the custom Turkey Point Technical Specifications.

By letter dated April 11, 1984 to J.P. O'Reilly Regional Administrator, Region II, FPL formalized commitments to implement the Turkey Point Performance Enhancement Program. It was the intention of FPL to review and implement, where appropriate, the philosophy and guidance of the Standard Technical Specifications in the development of upgraded plant procedures. In addition FPL committed to incorporate the requirements of the Standard Technical Specifications in all future proposed amendments to the Turkey Point Technical Specifications. NRC Confirmatory Order EA-84-55 dated July 13, 1984 required implementation of the Turkey Point Performance Enhancement Program (Revision I) and the commitments outlined in the April 11, 1984 letter.

In September of 1984 FPL voluntarily expanded the original commitment to include the development of a fully revised and reformated set of Turkey Point Technical Specifications within certain limitation. The revised set of Technical Specifications were to be based on Draft Revision 5 of NUREG 0452. The preparation limitations were that the revised Technical Specifications would not require hardware changes, would reflect the current Turkey Point plant design and analytical basis, and would consider operating hardship or reasonable resource additions. The Technical Specification Revision Project became Performance Enhancement Project (PEP) No. 10, an addition to the original PEP Projects 1 through 9 which were under the Confirmatory Order.

2.0 EVALUATION

The proposed amendment is a total replacement of the Turkey Point Units 3 and 4 current Technical Specifications with revised Technical Specifications which include the format and Guidance of the Standard Technical Specifications within the limitations discussed above.

2.0 Continued

Paragraph 50.36 of Part 10 of the Code of Federal Regulations requires that each licence authorizing operation of a commercial nuclear power plant shall include Technical Specifications that include the following category of information:

- Safety Limits, Limiting Safety System Settings and Limiting Control Settings
- Limiting Condition for Operation
- Surveillance Requirements
- Design Features
- Administrative Controls including reporting requirements

Although the current Technical Specifications include these categories of information the revised Technical Specifications will allow incorporation of additional information that has been gained through industry experience and incorporated in the Standard Technical Specifications. The revised Technical Specifications also include the format of the Standard Technical Specifications which has gained industry acceptance and will help resolve minor instances of uncertainty that may exist in the current Technical Specifications. The discussion that follows provides a general overview of how the required categories of information has improved in content, format and understandablity in the revised Technical Specifications. Appendix A is the Technical Specification Improvement Matrix which provides a summary of the improvement areas on an individual revised Technical Specification bases. Appendix B are the No Significant Hazards Evaluations for each revised Technical Specification and includes a specific summary and justification of changes for each revised Technical Specification.

2.1 SAFETY LIMITS, LIMITING SAFETY SYSTEM SETTINGS AND LIMITING CONTROL SETTING

Safety limits for nuclear reactors are limits upon important process variables which are found to be necessary to reasonably protect the integrity of certain of the physical barriers which guard against the uncontrolled release of radioactivity. Limiting Safety System settings for nuclear reactors are settings for automatic protective devices related to those variables having significant safety functions. Key safety limits and safety system settings are found primarily in Chapter 2.0 of both the current and revised Technical specifications. Revised Technical Specification improvements in Chapter 2.0 include the addition of explicit ACTION statements for Sections 2.1.1 Safety Limits - Reactor Core and Section 2.1.2 Safety

2.1 Continued

Limits - Reactor Coolant System Pressure and Reactor Trip System Instrumentation Setpoints. The revised Chapter 2.0 Technical Specifications have been revised to clearly indicate applicable modes consistent with the Standard Technical Specifications. Revised Technical Specification Section 2.2.1 includes a more complete set of trip functions. All of Section 2.0 revised Technical Specifications are reformatted in accordance with the Standard Technical Specifications.

2.2 <u>LIMITING CONDITIONS FOR OPERATION</u>

The limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When the Limiting Condition for Operation is not met, the licensee must shut down the reactor or follow remedial actions as indicated in the Technical Specifications.

The text format of Chapter 3.0 of the current Technical Specifications can make difficult, in some cases, the determination of the Limiting Condition of Operation, the required actions and plant operating modes for which conditions and actions are applicable. The revised Technical Specification uses the Standard Technical Specification format of explicit LIMITING CONDITIONS FOR OPERATION (LCO), Mode APPLICABILITY and ACTION statements for each Chapter 3.0 technical specification.

For some Chapter 3 current Technical Specifications the user must refer to specification 3.0 which provides a generic action requirement in instances where the ACTION statement is not explicitly stated for the individual Technical Specifications. The inclusion of an explicit ACTION statement in each Chapter 2.0 and 3.0 revised Technical Specification will provide the user with improved clarity and direction.

The Chapter 2.0 and 3.0 current Technical Specifications do not consistently use the current industry accepted practice of using the mode applicability numbers to define the plant operating mode for which LIMITING CONDITIONS or OPERATION and ACTION statements are applicable. The revised Technical Specifications provide an explicit mode APPLICABILITY statement for each Chapter 2.0 and 3.0 specification in accordance with the format of the Standard Technical Specifications.

2.2 Continued

In addition to these format changes to improve operator understanding and interpretation, the revised Technical Specifications provide additional limitations, restrictions and controls. Many of the revised Techical Specifications include more restrictive or additional LIMITING CONDITIONS FOR OPERATION and ACTION STATEMENTS. Appendix A, Improvement Matrix indicates which revised Technical Specifications are more restrictive than the current Technical Specifications because they contain additional or more restrictive LIMITING CONDITION FOR OPERATION, Mode APPLICABILITY or ACTION statements.

2.3 SURVEILLANCE REQUIREMENTS

The SURVEILLANCE REQUIREMENTS are requirements for tests, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within the safety limits, and that the limiting conditions of operation will be met.

Chapter 4.0 of the current and revised Technical Specifications provide the surveillance requirements. In the current Technical Specifications the Chapter 4 requirements are separated from their associated Chapter 3 system related LCO, APPLICABILITY and ACTION statements. In addition the surveillance requirements for a particular system may appear in two or more Chapter 4 locations. These features of the current Technical Specifications can make difficult the locating and identification of all LCO, APPLICABILITY, ACTION and SURVEILLANCE statements for a particular system. The revised Technical Specifications utilize the STS format of system Technical Specification which bring together the Chapter 3/4 LCO, APPLICABILITY, ACTION and SURVEILLANCE statements. One of the most significant improvements is that a majority of the revised Technical Specifications include added or more restrictive Surveillance Requirements (see Appendix A). Although most of the added surveillance requirements could previously be found in existing plant logs, procedures or testing programs, inclusion in the revised Technical Specifications bring these surveillance requirements into step with current industry requirements and practice.

2.4 DESIGN FEATURES

Design features are those features of the facility such as materials of construction and geometric arrangements, which if altered or modified, would have a significant effect on safety.

2.4 Continued

Chapter 5 provides a listing of design features in both the current and revised Technical Specifications. Five new Chapter 5 Technical Specifications have been added to the revised Technical Specifications consistent with the Standard Technical Specifications. Several Chapter 5 Technical Specifications are revised to provide additional information (See Appendix A).

2.5 ADMINISTRATIVE CONTROLS AND REPORTING REQUIREMENTS

Administrative controls are the provisions relating to organization and management, procedures, recordkeeping, review and audit and reporting necessary to assure operation of the facility in a safe manner.

Chapter 6 provides Administrative Controls in both the current and revised Technical Specifications. The Chapter 6 revised Technical Specifications are provided in a format consistent with the Standard Technical Specifications. Chapter 6 Technical Specifications include additional or more restrictive requirements than the current Technical Specifications (See Appendix A).

3.0 RELAXATIONS

Selected revised Technical Specifications contain relaxations from the current Technical Specifications. Many of the relaxations concern changes in the ACTION statement times or surveillance frequencies. In all but one case these relaxations bring the revised Technical Specifications in line with industry practice and the Standard Technical Specifications. The relaxations are justified in detail in the individual No Significant Hazards Evaluations.

4.0 CONCLUSION

The revised Technical Specifications represent a significant improvement in format, content and understandability for the Turkey Point operators and support groups. The Standard Technical Specification format will provide the NRC onsite inspectors and FPL engineering support groups a set of Technical Specifications that are consistent with the industry and, therefore, easier to use and locate information. As indicated in Section 3.0 and the attached Improvement Matrix, there is a significant increase in the content of information. The added or more restrictive LCO's, APPLICABILITY modes, ACTION and SURVEILLANCE statements will provide an increase in the margin of safety for system readiness and operation.

The standards used to arrive at a proposed determination that the proposed changes involve no significant hazards consideration are included in 10 CFR 50.92. The individual no Significant Hazards

4.0 Continued

Evaluations for each revised Technical Specification (attached Appendix B) demonstrate that the revised Technical Specifications do not involve an unreviewed safety question in that the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety.

Given these considerations there is reasonable assurance that the health and safety of the public will not be endangered by the proposed Turkey Point Units 3 and 4 Revised Technical Specifications.

APPENDIX A

TURKEY POINT UNITS 3 AND 4

REVISED TECHNICAL SPECIFICATIONS

IMPROVED MATRIX

TECHNICAL SPECIFICATIONS IMPROVEMENT MATRIX

TECH. SPEC. NUMBER	TITLE	IMPROVED FORMAT	NEW TECH.SPEC.	IMPROVED/ADDITIONAL/MORE RESTRICTIVE LCO	APPLICABILITY MODES	ACTION STATEMENT	SURV. REQMNTS	OTHER
1.0	DEFINITIONS	X						X
2.1	SAFETY LIMITS							
2.1.1	Reactor Core	X				X		X
2.1.2	Reactor Coolant System Pressure	X			X	X		X
2.2.1	Reactor Trip System Inst. Setpoints	X			X	X		X
3/4.0	APPLICABILITY	X						X
3/4.1	Reactivity Control Systems							
3/4.1.1	Boration Control							
3/4.1.1.1	Shutdown Margin-T_{avg} Greater Than 200 F	X			X	X	X	
3/4.1.1.2	Shutdown Margin-T_{avg} Less Than or Equal to 200 F		X					
3/4.1.1.3	Moderator Temperature Coefficient						X	
3/4.1.1.4	Minimum Temperature for Criticality		X					
3/4.1.2	Boration Systems							
3/4.1.2.1	Flow Path-Shutdown	X				X	X	
3/4.1.2.2	Flow Paths-Operating	X			X		X	
3/4.1.2.3	Charging Pump-Shutdown		X					
3/4.1.2.4	Charging Pumps-Operating	X			X		X	
3/4.1.2.5	Borated Water Source-Shutdown	X			X	X	X	
3/4.1.2.6	Borated Water Sources-Operating	X			X	X	X	
3/4.1.2.7	Heat Tracing	X			X		X	
3/4.1.3	Movable Control Assemblies							
3/4.1.3.1	Group Height	X		X	X	X		
3/4.1.3.2	Position Indication Systems-Operating	X		X	X	X		
3/4.1.3.3	Position Indication Systems-Shutdown	X		X	X	X	X	
3/4.1.3.4	Rod Drop Time	X		X				
3/4.1.3.5	Shutdown Rod Insertion Limit	X			X	X	X	
3/4.1.3.6	Control Rod Insertion Limits	X			X	X	X	
3/4.2	Power Distribution Limits							
3/4.2.1	Axial Flux Difference	X				X	X	
3/4.2.2	Heat Flux Hot Channel Factor	X						
3/4.2.3	Nuclear Enthalpy Rise Hot Channel Factor	X				X		
3/4.2.4	Quadrant Power Tilt Ratio	X		X		X	X	X
3/4.2.5	DNB Parameters	X		X			X	
3/4.3	Instrumentation							
3/4.3.1	Reactor Trip System Instrumentation	X		X	X			
3/4.3.2	Engineered Safety Features Actuation Sys. Instrumentation	X		X		X	X	

TECHNICAL SPECIFICATIONS IMPROVEMENT MATRIX

TECH. SPEC. NUMBER	TITLE	IMPROVED FORMAT	NEW TECH.SPEC.	IMPROVED/ADDITIONAL/MORE RESTRICTIVE				OTHER
				LCO	APPLICABILITY MODES	ACTION STATEMENT	SURV. REQMNTS	
3/4.3.3	Monitoring Instrumentation							
3/4.3.3.1	Radiation Monitoring For Plant Oper.	X		X	X	X	X	
3/4.3.3.2	Movable Incore Detectors			X	X	X	X	
3/4.3.3.3	Seismic Instrumentation			X	X	X	X	
3/4.3.3.4	Meteorological Instrumentation		X					
3/4.3.3.5	Accident Monitoring Instrumentation	X		X			X	X
3/4.3.3.6	Radioactive Liqd. Eff. Monitoring Inst.	X				X		
3/4.3.3.7	Radioactive Gaseous Eff. Monitoring Inst.		X					
3/4.3.4	Turbine Overspeed Protection			X	X	X	X	
3/4.4	Reactor Coolant System							
3/4.4.1.1	Startup and Power Operation	X			X			
3/4.4.1.2	Hot Standby	X		X				
3/4.4.1.3	Hot Shutdown	X		X				
3/4.4.1.4.1	Cold Shutdown-Loops Filled	X		X	X			
3/4.4.1.4.2	Cold Shutdown-Loops Not Filled	X		X	X			
3/4.4.2.1	Safety Valves-Shutdown	X		X	X	X	X	
3/4.4.2.2	Safety Valves-Operating	X		X		X	X	
3/4.4.3	Pressurizer	X		X		X	X	
3/4.4.4	Porv Block Valves	X					X	
3/4.4.5	Steam Generators	X				X	X	
3/4.4.6.1	RCS - Leakage Detection Systems	X		X	X	X	X	
3/4.4.6.2	RCS - Operational Leakage	X				X	X	
3/4.4.7	Chemistry	X				X		
3/4.4.8	Specific Activity	X		X		X	X	X
3/4.4.9.1	Press. Temp. Limits-Reactor Coolant Sys.	X				X	X	
3/4.4.9.2	Press. Temp. Limits-Pressurizer	X				X	X	
3/4.4.9.3	Press. Temp. Limits-Overpressure Prot.Sys.	X				X	X	
3/4.4.10	Structural Integrity	X		X		X		
3/4.4.11	Reactor Coolant System Vents	X					X	
3/4.5	Emergency Core Cooling Systems							
3/4.5.1	Accumulators	X		X	X	X	X	
3/4.5.2	ECCS Subsystems-T_{avg} Greater than or Equal to 350 F	X			X	X	X	
3/4.5.3	ECCS Subsystems-T_{avg} Less than 350 F		X					
3/4.5.4	Refueling Water Storage Tank	X		X	X		X	

TECHNICAL SPECIFICATIONS IMPROVEMENT MATRIX

TECH. SPEC. NUMBER	TITLE	IMPROVED FORMAT	IMPROVED/ADDITIONAL/MORE RESTRICTIVE					OTHER
			NEW TECH.SPEC.	LCO	APPLICABILITY MODES	ACTION STATEMENT	SURV. REQMNTS	
3/4.6	Containment Systems-Primary Containment							
3/4.6.1.1	Containment Integrity	X					X	
3/4.6.1.2	Containment Leakage	X		X	X	X	X	
3/4.6.1.3	Containment Air Locks	X		X			X	
3/4.6.1.4	Internal Pressure	X					X	
3/4.6.1.5	Air Temperature		X					
3/4.6.1.6	Containment Vessel Structural Integrity	X		X		X	X	
3/4.6.1.7	Containment Ventilation System	X		X	X	X	X	
3/4.6.2.1	Containment Spray System	X				X		
3/4.6.2.2	Containment Cooling System	X				X		
3/4.6.3	Emergency Containment Filtering System	X			X		X	
3/4.6.4	Containment Isolation Valves	X		X			X	
3/4.6.5	Combustible Gas Control Monitors	X			X		X	
3/4.6.6	Post Accident Containment Vent System	X					X	
3/4.7	Plant Systems - Turbine Cycle							
3/4.7.1.1	Safety Valves	X		X		X		
3/4.7.1.2	Auxiliary Feedwater System	X						
3/4.7.1.3	Condensate Storage Tanks	X						
3/4.7.1.4	Specific Activity	X		X		X	X	
3/4.7.1.5	Main Steam Line Isolation Valves	X				X	X	
3/4.7.1.6	Standby Feedwater System		X					
3/4.7.2	Steam Generator Pressure/Temp. Limitation		X					
3/4.7.3	Component Cooling Water System	X		X	X		X	
3/4.7.4	Intake Cooling Water System	X			X		X	
3/4.7.5	Ultimate Heat Sink		X					
3/4.7.6	Flood Protection		X					
3/4.7.7	Control Room Emergency Air Cleanup	X			X	X	X	
3/4.7.8	Snubbers	X		X				
3/4.7.9	Sealed Source Contamination	X		X			X	
3/4.8	Electrical Power Systems							
3/4.8.1.1	A.C. Sources - Operating							NOTE(1)
3/4.8.1.2	A.C. Sources - Shutdown							NOTE(1)
3/4.8.2.1	D.C. Sources - Operating							NOTE(1)
3/4.8.2.2	D.C. Sources - Shutdown							NOTE(1)
3/4.8.3.1	Onsite Power Dist.-Operating							NOTE(1)
3/4.8.3.2	Onsite Power Dist. - Shutdown							NOTE(1)

TECHNICAL SPECIFICATIONS IMPROVEMENT MATRIX

TECH. SPEC. NUMBER	TITLE	IMPROVED FORMAT	IMPROVED/ADDITIONAL/MORE RESTRICTIVE					OTHER
			NEW TECH.SPEC.	LCO	APPLICABILITY MODES	ACTION STATEMENT	SURV. REQMNTS	
3/4.9	Refueling Operations							
3/4.9.1	Boron Concentration	X				X		
3/4.9.2	Instrumentation	X			X		X	
3/4.9.3	Decay Time	X					X	
3/4.9.4	Containment Building Penetrations	X					X	
3/4.9.5	Communications	X					X	
3/4.9.6	Manipulator Crane	X		X		X	X	
3/4.9.7	Crane Travel-Spent Fuel Storage Areas	X		X			X	
3/4.9.8	Residual Heat Removal and Coolant Circ.							
3/4.9.8.1	High Water Level	X		X			X	
3/4.9.8.2	Low Water Level	X		X			X	
3/4.9.9	Containment Ventilation Isolation Sys.	X					X	
3/4.9.10	Water Level-Reactor Vessel		X					
3/4.9.11	Water Level-Storage Pool		X					
3/4.9.12	Handling of Spent Fuel Cask	X					X	
3/4.9.13	Radiation Levels Monitoring	X						
3/4.9.14	Spent Fuel Storage	X				X		
3/4.10	Special Test Exceptions							
3/4.10.1	Shutdown Margin	X		X				
3/4.10.2	Group Height, Insertion, and Power	X		X				
	Distribution Limits	X		X				
3/4.10.3	Physics Tests	X						
3/4.10.4	Reactor Coolant Loops	X						
3/4.10.5	Position Indication System-Shutdown		X					
3/4.11	Radioactive Effluents							
3/4.11.1.1	Liquid Effluents - Concentration	X					X	
3/4.11.1.2	Liquid Effluents - Dose	X						
3/4.11.1.3	Liquid Radwaste Treatment System	X						
3/4.11.2.1	Gaseous, Effluents - Dose Rate	X						
3/4.11.2.2	Dose - Noble Gases	X						
3/4.11.2.3	Dose - Iodine-131, Iodine-133, Tritium, and Radioactive	X						
3/4.11.2.4	Gaseous Radwaste Treatment System	X						
3/4.11.2.5	Explosive Gas Mixture	X						
3/4.11.2.6	Gas Decay Tanks	X					X	
3/4.11.3	Solid Radioactive Wastes	X						
3/4.11.4	Total Dose	X					X	
3/4.12	Radiological Enviornmental Monitoring							
3/4.12.1	Monitoring Program	X						
3/4.12.2	Land Use Census	X						
3/4.12.3	Interlaboratory Comparison Program	X						

TECHNICAL SPECIFICATIONS IMPROVEMENT MATRIX

TECH. SPEC. NUMBER	TITLE	IMPROVED FORMAT	IMPROVED/ADDITIONAL/MORE RESTRICTIVE: NEW TECH.SPEC.	LCO	APPLICABILITY MODES	ACTION STATEMENT	SURV. REQMNTS	OTHER
5.1	SITE	X						
5.1.1	Exclusion Area	X						
5.1.2	Low Population Zone							
5.1.3	Maps Defining Unrestricted Areas and Site Boundary for Radioactive Gaseous and Liquid Eff.	X						X
5.2	Containment							
5.2.1	Configuration		X					X
5.2.2	Design Pressure and Temperature	X						
5.2.3	Structural Loads	X						
5.2.4	Penetrations	X						
5.2.5	Containment Systems	X						
5.2.6	Containment Function	X						
5.3	Reactor Core							
5.3.1	Fuel Assemblies	X						
5.3.2	Control Rod Assemblies	X						X
5.3.3	Burnable Poison Rod Assemblies	X						X
5.4	Reactor Coolant System							
5.4.1	Design Pressure and Temperature					X		
5.4.2	Volume					X		
5.4.3	Seismic Stress							
5.5	Meteorological Tower Location		X					
5.6	Fuel Storage							
5.6.1	Criticality	X						X
5.6.2	Drainage		X					
5.6.3	Capacity		X					
5.6.4	Structure	X						
5.7	Component Cyclic or Transient Limit		X					

TECHNICAL SPECIFICATIONS IMPROVEMENT MATRIX

TECH. SPEC. NUMBER	TITLE	IMPROVED FORMAT	IMPROVED/ADDITIONAL/MORE RESTRICTIVE					OTHER
			NEW TECH.SPEC.	LCO	APPLICABILITY MODES	ACTION STATEMENT	SURV. REQMNTS	
6.1	Responsibility	X						X
6.2	Organization							
6.2.1	Offsite	X						X
6.2.2	Facility Staff	X						X
6.2.3	Shift Technical Advisor	X						
6.3	Facility Staff Qualifications	X						X
6.4	Training	X						X
6.5	Review and Audit							
6.5.1	Plant Nuclear Safety Committee	X						X
6.5.2	Company Nuclear Review Board (CNRB)	X						X
6.6	Reportable Event Action	X						
6.7	Safety Limit Violation	X						X
6.8	Procedures and Programs	X						X
6.9	Reporting Requirements							
6.9.1	Routine Reports	X						X
6.9.2	Special Reporting Requirements	X						
6.10	Record Retention	X						X
6.11	Radiation Protection Program	X						
6.12	High Radiation Area	X						X
6.13	Process Control Program (PCP)	X						X
6.14	Offsite Dose Calculation Manual (ODCM)	X						
6.15	Major Changes to Liquid, Gaseous, and Solid Radwaste Treatment Systems		X					

NOTES: (1) Technical Specification to be provided in a later submittal.
(2) The Standard Technical Specifications for Loose-Part Detection System and Motor-Operated Valve Thermal Overload Protection are not included in the revised Technical Specifications. These Technical Specifications are also not addressed in the current Turkey Point Technical Specification. The requirements of these Technical Specifications do not contribute to the primary success path of the plant safety analyses.

APPENDIX B

TURKEY POINT UNITS 3 AND 4

REVISED TECHNICAL SPECIFICATIONS

NO SIGNIFICANT HAZARDS EVALUATIONS

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: DEFINITION

NO: 1.0

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 1.0.

2) Proposed Condition of License:

a. The amendment reformats the definitions used in current Technical Specification into this specification for consistency with Standard Technical Specification definitions.

Additional definitions included in this revision that are not specified in the current Technical Specification are:

1. ACTUATION LOGIC TEST
2. AXIAL FLUX DIFFERENCE
3. FREQUENCY NOTATION
4. IDENTIFIED LEAKAGE
5. MASTER RELAY TEST
6. PRESSURE BOUNDARY LEAKAGE
7. SHUTDOWN MARGIN
8. SLAVE RELAY TEST
9. SOLIDIFICATION
10. SOURCE CHECK
11. STAGGERED TEST BASIS
12. TRIP ACTUATION DEVICE OPERATIONAL TEST
13. UNIDENTIFIED LEAKAGE
14. REFERENCE POSITION

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature.

1) The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which consolidates current requirements into a technical specification format consistent with the Standard Technical Specifications and does not involve technical or plant modifications.

Based on the above considerations the changes included in the development of proposed Technical Specification 1.0 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: SAFETY LIMITS - REACTOR CORE

NO: 2.1.1

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 2.1, and B2.1.

2) Proposed Condition of License:

a. The amendment reformats the current Technical Specification requirements into this Specification.

b. The revision is more complete than the current requirements as follows:

1. An ACTION statement is added for consistency with the Standard Technical Specification. The ACTION statement defines time limits for corrective actions (one hour) and references the appropriate ADMINISTRATIVE CONTROLS section (6.7.1) which contains reporting requirements and restrictions on unit operation (NRC authorization required prior to startup).

2. The current requirements for TWO and ONE Loop operation, and natural circulation are deleted because the current Turkey Point Safety Analysis does not allow power operation (MODES 1, and 2) with less than three loops.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The changes in Item 2.a are similar to example (i) of 48 FR 14870 in that they are administrative changes which consolidate current requirements into a Technical Specification format consistent with the Standard Technical Specifications and do not involve technical or plant modifications.

2) The changes in Items 2.b.1 and 2.b.2 are similar to example (ii) of 48 FR 14870 in that they provide additional restrictions and controls. The ACTION statement requires a mode reduction to HOT STANDBY or lower within one hour if the SAFETY LIMIT is violated. The SAFETY LIMIT applicability is also restricted to three loops in operation, consistent with safety analysis assumptions.

Based on the above considerations the changes included in the development of proposed Technical Specification 2.1.1 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: SAFETY LIMITS - REACTOR COOLANT SYSTEM PRESSURE

NO: 2.1.2

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 2.2 and B2.2.

2) Proposed Condition of License:

a. The amendment reformats the current Technical Specification requirements into this Specification.

b. The revision is more complete than the current requirements as follows:

An ACTION statement is added that requires plant shutdown within 1 hour and compliance with the applicable ADMINISTRATIVE CONTROLS Section 6.7.1 if the SAFETY LIMIT is not met in MODE 1 or 2. (Section 6.7.1 contains reporting requirements and restrictions on unit operation (NRC authorization required prior to startup).

In MODES 3, 4 or 5 the SAFETY LIMIT is required to be restored within 5 minutes and the applicable ADMINISTRATIVE CONTROLS Section 6.7.1 complied with.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The changes in Item 2.a are similar to example (i) of 48 FR 14870 in that they are administrative changes which consolidate current requirements into a Technical Specification format consistent with the Standard Technical Specifications and do not involve technical or plant modifications.

2) The change in Item 2.b is similar to example (ii) of 48 FR 14870 in that it provides additional restrictions and controls. The ACTION statement requires a mode reduction to HOT STANDBY or lower and the restoration of the SAFETY LIMIT within a fixed time period.

Based on the above considerations the changes included in the development of proposed Technical Specification 2.1.2 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

NO: 2.2.1

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 2.3.

2) Proposed Condition of License:

a. The amendment reformats the current requirements into the Standard Technical Specification format and explicitly states the APPLICABLE MODES and ACTION limits.

b. The revision is more complete than the current Technical Specification as follows:

1. The reactor trip setpoint table includes a more complete set of trip functions.

2. The MODE APPLICABILITY and ACTION requirements for each channel are explicitly stated.

c. The revision relaxes the following current requirements:

The trip setpoint table includes an allowance for channel drift.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement. Example (vi) relates to a change which may either result in some increase to the probability or consequences of a previously analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan: for example, a change resulting from the application of a small refinement of a previously used calculational model or design method.

1) The changes in Item 2.a are similar to example (i) of 48 FR 14870 in that they are administrative changes which consolidate current requirements into a Technical Specification format consistent with the Standard Technical Specifications and do not involve technical or plant modifications.

2) The changes in Items 2.b.1 and 2.b.2 are similar to example (ii) of 48 FR 14870 in that they provide a more complete set of trip functions and explicit MODE APPLICABILITY and ACTION limits are stated for each trip function. The more complete set of trip functions includes the Source and Intermediate Range trip channels required to be operable in Mode 2 and the steam generator water level low coincident with a steam flow/feed flow mismatch trip. The ACTION statement restrictions include appropriate power restrictions or increased surveillance requirements to compensate for a specific inoperable channel. For example, with an inoperable power range neutron flux channel, the ACTION statement restrictions include a power reduction and a high flux setpoint reduction or a quadrant power tilt surveillance frequency increase from 7 days to 12 hours.

3) The proposed change in item 2.c is similar to example (vi) of 48 FR 14870 in that it allows a specific allowance for setpoint drift between trip channel surveillance tests. This proposed change does not involve a significant hazards consideration because this change would not:

   a. Involve a significant increase in the probability of or consequences of an accident previously evaluated. The drift allowance in the proposed change is equal to or smaller than the drift allowance assumed in the safety analysis. These drift allowances have been used generically in Westinghouse plant safety analyses and experience has shown that the trip system instrumentation will perform within these drift tolerances.

Past operating experience at Turkey Point also indicates that the plant's trip system instrumentation drift will be less than the drift allowance. Therefore, an explicit allowance for drift in the trip setpoint table will not result in a significant increase in the probability of or consequences of an accident previously evaluated.

b. Create the possibility of a new or different kind of accident from any previously analyzed because the proposed change introduces no new mode of plant operation nor involves a physical modification to the plant.

c. Involve a significant reduction in a margin of safety because the proposed change does not involve changes in plant design, mode of plant operation or affect any safety analysis assumption.

Based on the above considerations the changes included in the development of proposed Technical Specification 2.2.1 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: APPLICABILITY

NO: 3/4.0

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.0, 4.0 and B3.0.

2) Proposed Condition of License:

a. The amendment consolidates the current Technical Specification requirement into this Specification for consistency with Standard Technical Specification.

b. The revision is more complete than the current Technical Specification as follows:

1. Places greater restriction on the surveillance grace period so as not to exceed 3.25 times the normal interval over 3 consecutive tests. The current Technical Specification allows 25% grace period on each interval.

2. Includes a statement that failure to perform the required surveillance test makes the component inoperable.

3. Includes generic statements incorporating the plant ISI program submitted to meet the intent of ASME Section XI. The current Technical Specification does not contain this requirement.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1. The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which consolidates current requirements into a technical specification format consistent with the Standard Technical Specifications and does not involve technical or plant modifications.

2) The proposed changes as described in Items 2.b.1, 2.b.2 and 2.b.3 are similar to example (ii) of 48 FR 14870 in that they provide additional controls by specifically limiting the accumulative use of the grace period and ISI compliance to requirements of ASME Section XI. Additional restriction is included by defining a missed surveillance test as cause for component inoperability.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.0 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: SHUTDOWN MARGIN, T_{AVG} GREATER THAN 200 DEGREES -F

NO: 3/4.1.1.1

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specifications 3.2.1.f, 3.2.4.c, Table 4.1-2, item 1.e., 4.11, and 6.9.2.a(4).

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this Specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more complete than the current Technical Specification as follows:

1. The SHUTDOWN MARGIN LCO applies to more operational modes, (Modes 3 and 4).

2. The ACTION statement includes minimum flow and boron concentration requirements.

3. In addition to boron concentration the SURVEILLANCE REQUIREMENTS include the following parameter surveillances: Control rod position, Reactor Coolant System temperature, Fuel Burnup and Xenon and Samarium concentration.

c. The revision relaxes the following current requirements:

1. In the event that a control rod is inoperable, the current Technical Specification requires boration to compensate for the inoperable control rod. The proposed Technical Specification requires boration to compensate for an untrippable inoperable control rod only if the required SHUTDOWN MARGIN is not met, with an increased allowance for the withdrawn worth of the inoperable rod.

2. The current Technical Specification requires RCS boron concentration surveillance twice per week. The proposed Technical Specification requires RCS boron concentration surveillance at least once per 31 EFPD's in MODES 1 and 2.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement. Example (vi) relates to a change which may either result in some increase to the probability or consequences of a previously analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan: for example, a change resulting from the application of a small refinement of a previously used calculational model or design method.

1) The changes in item 2.a are similar to example (i) of 48 FR 14870 in that they are administrative changes which consolidate current requirements into a Technical Specification format consistent with the Standard Technical Specifications and do not involve technical or plant modifications.

2) The changes in items 2.b.1, 2.b.2 and 2.b.3 are similar to example (ii) of 48 FR 14870 in that they provide additional restrictions and controls in the added MODES and the more complete and restrictive ACTION and SURVEILLANCE REQUIREMENTS. The MODE applicability is expanded to include SHUTDOWN MODES 3 and 4. The ACTION statement includes minimum boron concentration and flow requirements when boration is required and the SURVEILLANCE REQUIREMENTS include daily SURVEILLANCE of boron concentration, control rod position, Reactor Coolant System temperature, Fuel Burnup, and Xenon and Samarium concentration in MODES 3 and 4.

3) The proposed changes in item 2.c.1 and 2.c.2 are similar to example (vi) of 48 FR 14870. The change in 2.c.1 involving boration to increase the SHUTDOWN MARGIN is only required if an inoperable rod would result in a SHUTDOWN MARGIN limit violation. The change in item 2.c.2 involves the RCS boron concentration surveillance frequency reduction in MODES 1 and 2 from twice weekly to once per 31 EFPD's. These changes do not involve a significant hazards consideration because the changes will not:

a. Involve a significant increase in the probability of or consequences of an accident previously evaluated.

The proposed change to require boration only when the SHUTDOWN MARGIN limit is violated as the result of an inoperable rod is sufficient to preserve the safety analysis assumption involving SHUTDOWN MARGIN requirements for accident mitigation. Because the accident analysis only assumes that the minimum SHUTDOWN MARGIN required by the Technical Specifications is available, preserving the limit, as required by the proposed Technical Specification, ensures that all accident analysis results that depend on SHUTDOWN MARGIN remain valid.

The proposed reduction in the RCS boron concentration surveillance in MODES 1 and 2 from twice per week to once per 31 EFPD's is adequate to support the SHUTDOWN MARGIN Technical Specification limit because the RCS boron concentration is not directly related to SHUTDOWN MARGIN in MODES 1 and 2. The SHUTDOWN MARGIN in MODES 1 and 2 is ensured by surveillance of the control rod bank position and verifying that the rod bank withdrawal is within the allowable withdrawal limit. The proposed Technical Specification surveillance frequency on rod bank position is once per 12 hours.

The 31 EFPD surveillance of RCS boron concentration is used in the overall core reactivity balance to demonstrate agreement with the predicted core reactivity trend over the fuel cycle. Past operating experience at Turkey Point as well as at other Westinghouse plants has shown that core reactivity trends change slowly with time and that the Standard Technical Specification surveillance frequency of once each 31 EFPD's is adequate for ensuring that the actual core depletion is following the predicted reactivity trend throughout the fuel cycle.

b. Create the possibility of a new or different kind of accident from any previously analyzed because the proposed change introduces no new mode of plant operation nor involves a physical modification to the plant.

c. Involve a significant reduction in a margin of safety because the proposed change does not involve changes in plant design, mode of plant operation or affect any safety analysis assumption.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.1.1.1 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: SHUTDOWN MARGIN - T_{AVG} LESS THAN OR EQUAL TO 200 DEGREES F

NO: 3/4.1.1.2

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

There is no corresponding LCO requirement in the current Turkey Point Technical Specifications. A related Surveillance Requirement is contained in Table 4.1-2, item 1.e.

2) Proposed Condition of License:

a. The amendment adds a new Technical Specification that specifies the LCO, APPLICABLE MODES, ACTION statements and SURVEILLANCE REQUIREMENTS for SHUTDOWN MARGIN in MODE 5 operation.

b. The revision is more restrictive than the current requirements as follows:

1. A new LCO is added which contains SHUTDOWN MARGIN requirements applicable to MODE 5.

2. An explicit ACTION statement is included which requires immediate boration of 10 gpm of 20,000 ppm boron or equivalent if an LCO violation occurs.

3. The SURVEILLANCE REQUIREMENTS include a daily determination of SHUTDOWN MARGIN including changes in SHUTDOWN MARGIN due to changes in RCS temperature, boron concentration, control rod position, fission product concentration and fuel burnup. The impact of an inoperable rod must also be included.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

Proposed Tech. Spec. No. 3/4.1.1.2

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The change in Item 2.a is similar to example (i) of 48 FR 14870 in that they are administrative changes which consolidate current requirements into a Technical Specification format consistent with the Standard Technical Specifications and do not involve technical or plant modifications.

2) The changes in Items 2.b.1, 2.b.2 and 2.b.3 are similar to example (ii) of 48 FR 14870 in that they provide a new LCO for controlling SHUTDOWN MARGIN in MODE 5, an ACTION statement requiring immediate boration if the LCO limit is violated, and a SURVEILLANCE REQUIREMENT to determine the SHUTDOWN MARGIN at least once every 24 hours.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.1.1.2 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: MODERATOR TEMPERATURE COEEFFICIENT (MTC)

NO: 3/4.1.1.3

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

   As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.1.2.1.

2) Proposed Condition of License:

   a. The amendment consolidates the current requirements into this specification and adds SURVEILLANCE REQUIREMENTS.

   b. The revision is more restrictive than the current requirements because:

      SURVEILLANCE REQUIREMENTS require measuring the MTC prior to initial power operation to ensure that the MTC is less positive than the positive limit. As the end of cycle is approached an MTC measurement is again required to ensure that the MTC is less negative than the negative limit.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The change in item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which consolidates current requirements into a Technical Specification format consistent with the Standard Technical Specifications and does not involve technical or plant modifications.

2) The change in item 2.b is similar to example (ii) of 48 FR 14870 in that it provides more complete and restrictive SURVEILLANCE REQUIREMENTS to ensure that the temperature coefficient is within its limit.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.1.1.3 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: MINIMUM TEMPERATURE FOR CRITICALITY

NO: 3/4.1.1.4

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

There is no corresponding requirement in the current Turkey Point Technical Specifications.

2) Proposed Condition of License:

a. The amendment adds a new Technical Specification that specifies the LCO, APPLICABLE MODES, ACTION statements and SURVEILLANCE REQUIREMENTS for minimum temperature for criticality.

b. The proposed change is more restrictive than the current requirements because it represents an explicit limit on the minimum temperature allowed for critical operation, required ACTIONS if a limit violation occurs and SURVEILLANCES to ensure that the limit is not exceeded. The ACTION statement allows 15 minutes to restore the temperature or be in HOT STANDBY and the SURVEILLANCE requires a temperature check within 15 minutes of achieving criticality.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

The changes in items 2.a and 2.b are similar to example (ii) of 48 FR 14870 in that they provide additional restrictions and controls by addition of a new Technical Specification for the MINIMUM TEMPERATURE FOR CRITICALITY. The new technical specification includes more complete and restrictive ACTION and SURVEILLANCE REQUIREMENTS. The ACTION statement allows 15 minutes to restore the temperature or be in HOT STANDBY and the SURVEILLANCE requires a temperature check within 15 minutes of achieving criticality.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.1.1.4 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: BORATION SYSTEMS FLOW PATH - SHUTDOWN

NO: 3/4.1.2.1

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.6.a, Table 4.1-1 Item 19 and Table 4.18-1, item 8.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this proposed specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more restrictive than the requirements as follows:

1. ACTION limits are more explicit,

2. SURVEILLANCE REQUIREMENTS include surveillance of flow path temperature.

c. The revision relaxes the following current requirement:

A test of the boric acid flow controller is deleted

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed changes as described in Item 2.a are similar to example (i) of 48 FR 14870 in that they are administrative changes which consolidate current requirements into a technical specification format consistent with the Standard Technical Specifications and do not involve technical or plant modifications.

2) The proposed changes as described in Items 2.b.1 and 2.b.2 are similar to example (ii) of 48 FR 14870 in that they provide additional restrictions and controls in the added ACTION limits, which requires that any CORE ALTERATION operation or positive reactivity addition be suspended if no boron injection flow path is OPERABLE. The surveillance requirements are more complete because they include a temperature surveillance with a minimum temperature requirement of 145 F on the heat traced portion of the flow path.

3) The proposed change to delete the test of the boric acid flow controller does not involve a significant hazards consideration because this change would not:

   a. Involve a significant increase in the probability of or consequences of an accident previously evaluated. The boric acid controller is not used by any automatic accident mitigation system. Therefore, the safety analysis is not affected by the operability of the boric acid controller. Also the boric acid controller is the normal method for varying RCS boron concentration. As such, the system is proven to be operable each time it is used. The current Technical Specification surveillance requirement to perform a surveillance test once each refueling provides an insignificant increase in the system reliability. Therefore, there is no significant increase in the probability of or consequences of an accident previously evaluated because the boric acid controller surveillance is deleted.

b. Create the possibility of a new or different kind of accident from any previously analyzed because the proposed change introduces no new mode of plant operation nor involves a physical modification to the plant.

c. Involve a significant reduction in a margin of safety because the proposed deletion of the boric acid flow controller does not involve changes in plant design, mode of plant operation or affect any safety analysis assumption.

The proposed technical specification deletion of the boric acid flow controller test is consistent with industry practices in that the Standard Technical Specifications do not include the subject test.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.1.2.1 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: BORATION SYSTEMS FLOW PATHS - OPERATING

NO: 3/4.1.2.2

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.6.a, 3.6.b.2 and b.4, 3.6.c.2 and c.4, 3.6.d.2, and 4.18, Table 4.18-1, item 8.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

An explicit SURVEILLANCE REQUIREMENT in the current requirements to verify the availability of power to required flow path components is included in the definition of OPERABILITY in the proposed change.

b. The revision is more complete than the current Technical Specification as follows:

1. APPLICABILITY REQUIREMENTS are more restrictive because they apply to more MODES, (Modes 3 and 4).

2. SURVEILLANCE REQUIREMENTS include surveillance of flow path temperature and boric acid pump flow rate.

c. The revision relaxes the following current requirements:

1. In the current specification the two required boron injection flow paths must be from the RWST and BAT. In the proposed specification the two required boron injection flow paths can be from the RWST and BAT or two flow paths can be from the RWST.

2. OPERABILITY requirements on the boric acid pumps are tied to the associated flow path OPERABILITY. A boric acid pump is only required to be operable when its associated flow path is required to be OPERABLE.

3. The allowed outage time for a boric acid or charging pump is relaxed from 24 hours to 72 hours.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed changes as described in Item 2.a are similar to example (i) of 48 FR 14870 in that they are administrative changes which consolidate current requirements into a technical specification format consistent with the Standard Technical Specifications and do not involve technical or plant modifications.

   The deletion of the explicit surveillance of the flow path component power source has no impact on flow path OPERABILITY because the Revised Technical Specification definition of OPERABILITY (Section 1.0) requires that all support functions needed by an OPERABLE component must also be capable of performing their related support functions.

2) The proposed changes as described in Items 2.b.1 and 2.b.2 are similar to example (ii) of 48 FR 14870 in that they provide boration system flow path restrictions applicable in MODES 3 and 4 and more complete surveillance requirements for determining flow path operability.

3) The proposed changes in item 2.c do not involve a significant hazards consideration because these changes do not:

a. Involve a significant increase in the probability of or consequences of an accident previously evaluated.

The proposed more flexible boron injection flow path requirement has no impact on previously evaluated accidents because these boron injection flow paths are not assumed to be operable for accident mitigation. The flow paths required to be operable for ESF accident mitigation are controlled by other Technical Specifications. Furthermore, the proposed flow path requirement contains the same number of required flow paths (i.e., 2) as tne current requirement and is consistent with industry practice in that the boration flow paths are those required in the Standard Technical Specification.

The boric acid pump OPERABILITY requirement deletes the requirement for a redundant OPERABLE boric acid pump. The current requirement for BAT PUMP OPERABILITY independent of the boron injection flowpath OPERABILITY requirement only serves to ensure that a redundant boric acid pump would be available if needed by an OPERABLE boron injection flow path. The current requirement then tends to improve the availability of the boration flow path. Because the boric acid pumps are not required to be OPERABLE for accident mitigation and are not actuated by the reactor trip or ESF actuation systems this availability reduction has no impact on any accident previously evaluated.

The boric acid pump and charging pump allowed outage time relaxation also impacts the flow path availability but represents no increase in the probability of or consequences of an accident previously evaluated because these components are not assumed to be operable for accident mitigation in the safety analysis. The proposed revision is consistent with industry practices in that the Standard Technical Specifications allow a 72 hour outage time for one boron injection flowpath inoperable.

b. Create the possibility of a new or different kind of accident from any previously analyzed because the proposed change introduces no new mode of plant operation nor involves a physical modification to plant.

c. Involve a significant reduction in a margin of safety because the above relaxations in the current requirements do not affect any system or component required to be operable for accident mitigation.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.1.2.2 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: BORATION SYSTEMS CHARGING PUMPS - SHUTDOWN

NO: 3/4.1.2.3

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

There is no corresponding requirement in the current Turkey Point Specifications.

2) Proposed Condition of License:

a. The amendment adds a new technical specification that specifies the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS for operability of charging pumps in Modes 5 and 6.

b. The revision is more complete than the current Technical Specification because it includes a specific ACTION statement which restricts operation involving CORE ALTERATIONS or positive reactivity changes with no OPERABLE charging pump. An explicit pump surveillance is also included.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed changes as described in Items 2.a and 2.b are similar to example (ii) of 48 FR 14870 in that they provide additional information by including the requirements for charging pump operability in Modes 5 and 6 in Standard Technical Specification format.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.1.2.3 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: BORATION SYSTEMS CHARGING PUMPS - OPERATING

NO: 3/4.1.2.4

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.6.b.1, 3.6.c.1, 3.6.d.1, Table 4.1-1, items 12 and 16.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more complete than the current Technical Specification as follows:

1. APPLICABILITY REQUIREMENTS are more restrictive because they apply to more MODES (MODES 3 and 4).

2. SURVEILLANCE REQUIREMENTS are more explicit. Entry into Mode 3 is allowed for surveillance testing, consistent with current plant practice, to allow a more meaningful test of the positive displacement pumps by testing with a higher RCS backpressure.

3. The LCO requires each OPERABLE charging pump to be powered from independent power supplies.

c. The revision relaxes the following current requirements:

1. The allowed outage time for the charging pumps is relaxed from 24 to 72 hours.

2. A SURVEILLANCE REQUIREMENT to calibrate the charging pump flow channel is deleted.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed changes as described in Item 2.a are similar to example (i) of 48 FR 14870 in that they are administrative changes which consolidate current requirements into a technical specification format consistent with the Standard Technical Specifications and do not involve technical or plant modifications.

2) The proposed changes as described in Items 2.b.1, 2.b.2 and 2.b.3 are similar to example (ii) of 48 FR 14870 in that they provide additional restrictions by including MODES 3 and 4 in the MODE APPLICABILITY requirements, by including an explicit SURVEILLANCE REQUIREMENT for the charging pumps and by requiring independent power supplies.

3) The proposed changes to relax the charging pump allowed outage time and the charging pump flow channel calibration does not involve a significant hazards consideration because these changes would not:

a. Involve a significant increase in the probability of or consequences of an accident previously evaluated. The longer outage time is supported by the system design which includes three charging pumps where only two charging pumps are required to be operable to meet the Technical Specification. The revised outage time is also consistent with the Standard Technical Specifications. In addition the charging pumps are not used for accident mitigation. Therefore the above proposed changes to relax the charging pump allowed outage time has no impact on any previously evaluated accident.

Deleting the calibration of the charging pump flow channel has no impact on any previously evaluated accident because the flow signal is not used for any automatic or manual accident industry practice mitigation function. This change is also consistent with industry practice in that the calibration is not required by the Standard Technical Specifications.

b. Create the possibility of a new or different kind of accident from any previously analyzed because the proposed change introduces no new mode of plant operation nor involves a physical modification to the plant.

c. Involve a significant reduction in a margin of safety because the proposed changes in the charging pump allowed outage time and flow channel calibration have no impact on any safety analysis assumption.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.1.2.4 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: BORATED WATER SOURCE - SHUTDOWN

NO: 3/4.1.2.5

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Table 4.1-1 Item 14 and Table 4.1.2 Item 3.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more complete than the current Technical Specification as follows:

1. The LCO represents a new borated water source LCO for MODES 5 and 6.

2. The ACTION statement identifies operating restrictions to suspend CORE ALTERATIONS or reactivity increases if the LCO is violated.

3. The SURVEILLANCE REQUIREMENTS are more complete and include surveillance of the liquid volume and temperature.

c. The revision relaxes the following current requirements:

1. BAT boron concentration surveillance is relaxed from twice weekly to weekly.

2. The BAT level instrument weekly CHANNEL CHECK is deleted.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which consolidates current requirements into a technical specification format consistent with the Standard Technical Specifications and do not involve technical or plant modifications.

2) The proposed changes as described in Items 2.b.1, 2.b.2 and 2.b.3 are similar to example (ii) of 48 FR 14870 in that they provide a BORATED WATER SOURCE LCO applicable to MODES 5 and 6, an ACTION statement to suspend CORE ALTERATIONS or positive reactivity changes if the LCO is not satisfied and SURVEILLANCE of BORATED WATER volume and temperature in addition to boron concentration.

3) The proposed changes to relax the BAT boron concentration surveillance frequency and the weekly BAT level instrument CHANNEL CHECK do not involve a significant hazards consideration because these changes would not:

   a. Involve a significant increase in the probability of or consequences of an accident previously evaluated. Past operating experience at Turkey Point indicates that surveillance of boron concentration in the BAT tanks once a week is adequate because boron concentration does not vary significantly over a one week time interval in MODES 5 and 6. This surveillance frequency is also consistent with industry practices as it is the required surveillance interval provided in the Standard Technical Specification. Therefore, surveillance of BAT boron concentration at a once per week interval will assure adequate boron concentration is available.

The deletion of the weekly BAT level instrument CHANNEL CHECK has minimal impact because the same CHANNEL CHECK is required in the proposed Accident Monitoring Instrument Technical Specification (3/4.3.3.5). The Technical Specification 3/4.3.3.5 surveillance requires a CHANNEL CHECK once per month in MODES 1 thru 3. The stable reactivity condition of the core in MODES 5 and 6 makes the need for boration unlikely. The weekly surveillance of the BAT liquid volume itself serves as a CHANNEL CHECK because the most probable channel failures result in instrument readings pegged at the upper or lower range limit of the instrument. Either of these readings would be a change from the expected reading and alert the operator of a potential instrument problem. In addition, the BAT is not required to be OPERABLE for accident mitigation by the reactor trip or ESF actuation system. The CHANNEL CHECK of the BAT level instrument as proposed in revised technical specification 3/4.3.3.5 is consistent with industry practice in that it is that required by the Standard Technical Specification.

Because the relaxed surveillances of the BAT boron concentration and level instrument CHANNEL CHECK have a minimal impact on the availability of the BORATED WATER SOURCE and the BAT is not required for accident mitigation, there is no significant increase in the probability of or consequences of an accident previously evaluated.

b. Create the possibility of a new or different kind of accident from any previously analyzed because the proposed change introduces no new mode of plant operation nor involves a physical modification to the plant.

c. Involve a significant reduction in a margin of safety because the BAT and Borated Water Service Sources are not required for accident previously evaluated. The weekly boron concentration surveillance is considered adequate as indicated above and the additional surveillances in the proposed technical specifications 3/4.1.2.5 and 3/4.3.3.5 more than compensate for the deletion of the weekly BAT level instrument CHANNEL CHECK. The proposed BAT boron concentration surveillance relaxation may result in a relatively minor change in the BORATED WATER SOURCE availability but the new LCO and ACTION statements in total provide additional assurance that the BORATED WATER SOURCE will be available in MODES 5 and 6 to borate the RCS.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.1.2.5 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: BORATION WATER SOURCES - OPERATING

NO: 3/4.1.2.6

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

   As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specifications 3.4.1.a.1, 3.6.b.3, 3.6.b.6, 3.6.c.3, 3.6.c.6, Table 4.1-1 Items 14 and 15, and Table 4.1-2 Items 2 and 3.

2) Proposed Condition of License:

   a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

   b. The revision is more complete than the current Technical Specification as follows:

      1. APPLICABILITY REQUIREMENTS are more restrictive because they apply to more modes (MODES 3 and 4).

      2. ACTION limits are more restrictive, with explicit time limits to restore LCO's. (72 hours for BAT, 1 hour for RWST).

      3. SURVEILLANCE REQUIREMENTS are more complete and include minimum liquid volume, and temperature in the BAT and RWST and a maximum RWST temperature.

   c. The revision relaxes the following current requirements:

      1. BAT boron concentration surveillance is relaxed from twice weekly to weekly.

      2. The BAT and RWST level instrument channel checks are relaxed from weekly to monthly.

      3. Primary water storage tank minimum volume requirement is deleted.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed changes as described in Item 2.a are similar to example (i) of 48 FR 14870 in that they are administrative changes which consolidate current requirements into a technical specification format consistent with the Standard Technical Specifications and do not involve technical or plant modifications.

2) The proposed changes as described in Items 2.b.1, 2.b.2 and 2.b.3 are similar to example (ii) of 48 FR 14870 in that the borated water source LCO is applied to more MODES (3 and 4), the ACTION limit identifies explicit time limits for restoring the LCO (72 hours for the BAT and 1 hour for the RWST), and the SUREVEILLANCE REQUIREMENTS include the liquid volume and temperature in addition to the boron concentration.

3) The proposed changes to relax the current requirements identified in 2.c above do not involve a significant hazards consideration because these changes will not:

   a. Involve a significant increase in the probability of or consequences of an accident previously evaluated.

The Boric Acid Tanks store sufficient boron to bring the unit to COLD SHUTDOWN and is a backup source of negative reactivity for the reactor trip and safety injection systems which are used for accident mitigation. The proposed SURVEILLANCE REQUIREMENT reduces the frequency of boron concentration surveillance from twice weekly to weekly, but adds a weekly surveillance of the boric acid volume and temperature. The proposed BAT SURVEILLANCE REQUIREMENTS therefore represent a more complete surveillance of the BAT as a source of borated water than the current requirements and in this sense are more restrictive.

Past operating experience at Turkey Point indicates that surveillance of boron concentration in the BAT tanks once a week is adequate because boron concentration does not vary significantly over a one week time interval. This surveillance frequency is also consistent with industry practices as it is the required surveillance interval provided in the Standard Technical Specifications. Therefore, surveillance of BAT boron concentration, boric acid volume, and temperature at a once per week interval will assure an adequate borated water source is available from the BAT.

The BAT and RWST level instrument channel checks are deleted in the proposed borated water sources Technical Specification but are included in the Accident Monitoring Instrument Technical Specification. The SURVEILLANCE REQUIREMENTS include a monthly channel check in MODES 1, 2 and 3 and a calibration once per refueling. The current Technical Specifications require a weekly channel check.

The RWST and BAT level channel check relaxation is justified because of the standby status of these components. Further, the weekly surveillance of the RWST and BAT liquid volume itself serves as a channel check because the most probable channel failures result in instrument readings pegged at the upper or lower range limit of the instrument. Either of these readings would be a change from the expected reading and alert the operator of a potential instrument problem. In addition, both high and low RWST and BAT level alarm annunciators are available to alert the operator of an abnormal level condition.

The primary water storage tank serves as a source of make-up water for the RCS but is not required to be OPERABLE for any safety function. Normally primary water is blended with boric acid to reduce the boron concentration prior to makeup injection into the RCS. However fully concentrated boric acid from the BAT can be used for makeup using the normal flow path through the boric acid blender or through the emergency boration flow path. The RWST can also be used for RCS make-up through the charging pumps.

The Standard Technical Specifications also do not require the primary water storage tank to be OPERABLE. The deletion of the primary water storage tank (PWST) Technical Specification is therefore justified because it is not assumed to be OPERABLE for accident mitigation in the safety analysis.

In summary, the relaxations of the current Technical Specification requirements identified in Item 2.c will not increase the probability of or consequences of an accident previously evaluated. The relaxations of the BAT and PWST requirements have no impact on the safety analysis because these components are not required to be OPERABLE for accident mitigation by the safety analysis. The BAT and RWST level instrument channel check relaxation has no impact on any previously evaluated accident because of the standby nature of these components, the weekly surveillance of the BAT and RWST water volume, and the presence of annunciators which will indicate abnormally high or low levels.

b. Create the possibility of a new or different kind of accident from any previously analyzed because the proposed change introduces no new mode of plant operation nor involves a physical modification to the plant.

c. Involve a significant reduction in a margin of safety. The proposed relaxations of the current BAT and PWST Technical Specification requirements do not reduce any safety margin because they have no impact on any safety analysis assumption. The relaxation of the RWST level instrument check surveillance will not significantly reduce any safety margin because of other SURVEILLANCE REQUIREMENTS and the presence of RWST level high and low alarms which will annunciate an abnormal RWST level condition.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.1.2.6 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: HEAT TRACING

NO: 3/4.1.2.7

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specifications 3.6.b.5, 3.6.c.5, 3.6.d, and 4.18, Table 4.18-1, Item 8.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more complete than the current Technical Specification as follows:

1. APPLICABILITY REQUIREMENTS are more restrictive because they apply to more MODES (MODE 3).

2. SURVEILLANCE REQUIREMENTS are more complete and include a temperature check of the heat traced lines.

c. The revision relaxes the following current requirement:

The allowed outage time for one channel of heat tracing is relaxed from 24 hours to 30 days provided a temperature check is done every 8 hours.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

Proposed Tech. Spec. No. 3/4.1.2.7

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed changes as described in Item 2.a are similar to example (i) of 48 FR 14870 in that they are administrative changes which consolidate current requirements into a technical specification format consistent with the Standard Technical Specifications and do not involve technical or plant modifications.

2) The proposed changes as described in Items 2.b.1 and 2.b.2 are similar to example (ii) of 48 FR 14870 in that they require two OPERABLE channels of heat tracing in MODE 3 as well as in MODES 1 and 2 as currently required and they include a daily surveillance of the boric acid flow path temperature.

3) The proposed change to relax the allowed outage time for one channel of heat tracing does not involve a significant hazards consideration because this change would not:

   a. Involve a significant increase in the probability of or consequences of an accident previously evaluated. The purpose of heat tracing the boric acid flow path is to ensure that the temperature of the boric acid in the flow path is maintained above its solubility limit of 135 F. One channel of heat tracing is sufficient for maintaining the temperature above this limit. Therefore the ACTION statement requirement to perform an 8 hour temperature surveillance when one heat trace channel is not OPERABLE is sufficient to ensure the heat tracing function is being performed. Also, the boric acid flow path is not required to be OPERABLE for accident mitigation in the safety analysis. Therefore the proposed allowed outage time relaxation involves no increase in the probability of or consequences of an accident previously evaluated.

b. Create the possibility of a new or different kind of accident from any previously analyzed because the proposed change introduces no new mode of plant operation nor involves a physical modification to the plant.

c. Involve a significant reduction in a margin of safety because the proposed relaxation in allowed outage time is countered by a required increase in a temperature surveillance frequency which ensures that the heat tracing function is being performed. Also, the proposed relaxation in allowed outage time is consistent with Industry practice in that 30 days is the allowed outage time for one channel of heat tracing in the Standard Technical Specifications. Therefore the proposed relaxation involves no significant reduction in a margin of safety.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.1.2.7 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: MOVABLE CONTROL - ASSEMBLIES - GROUP HEIGHT

NO: 3/4.1.3.1

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specifications 3.2.2, 3.2.4a, 3.2.4b, 3.2.5, and Table 4.1-2 Item 5.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more restrictive than the current Technical Specification as follows:

1. Continued power operation with an inoperable rod is only permitted if the rod is trippable.

2. APPLICABILITY REQUIREMENTS are more restrictive because they apply to more MODES (MODE 2).

3. The ACTION time limit to restore the LCO is more restrictive (1 hour vs 8 hours)..

c. The revision relaxes the following current requirements.

1. The reactivity limit for continued power operation of 0.3% of the maximum worth of an ejected inoperable rod is deleted.

2. The requirement to reduce the hi-flux trip setpoint when both rod deviation and power range channel deviation alarms are inoperable is deleted.

3. Surveillance intervals for determining OPERABLE rods are relaxed from 14 days to 31 days.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed changes as described in Item 2.a are similar to example (i) of 48 FR 14870 in that they are administrative changes which consolidate current requirements into a technical specification format consistent with the Standard Technical Specifications and do not involve technical or plant modifications.

2) The proposed changes as described in Items 2.b.1, 2.b.2 and 2.b.3 are similar to example (ii) of 48 FR 14870 in that added restrictions are placed on continued power operation with an INOPERABLE control rod. Continued power operation is allowed only if the rod is still able to perform its safety function (i.e. trippable). The proposed changes also are more restrictive because they apply to more modes (MODE 2) and the ACTION time allowed to evaluate a potential inoperable rod condition is shorter (1 hour vs. 8 hours).

3) The proposed change to relax the requirements in Item 2.c above, does not involve a significant hazards consideration because these changes do not:

   a. Involve a significant increase in the probability of or consequences of an accident previously evaluated.

The reactivity limit in the current Technical Specification is not needed to preserve any rod ejection analysis design assumption. Other restrictions in the proposed Technical Specification for MOVABLE CONTROL ASSEMBLIES ensure that the normal rod insertion and alignment limits are preserved thereby preserving the original safety analysis limiting assumptions related to rod position.

The current Technical Specification requirement to reduce the hi-flux trip setpoint when both the rod deviation and the power range channel deviation alarm are inoperable is replaced with more frequent surveillances. The rod position surveillance is increased from once per 12 hours to once per 4 hours when the rod deviation alarm is inoperable. In the quadrant power tilt Technical Specification, the power tilt surveillance is increased from once per 7 days to once per 12 hours when the power range deviation alarm is inoperable. These increased surveillances will adequately compensate for an INOPERABLE rod position deviation or power range channel deviation alarm and are consistent with industry practice in that these are the same SURVEILLANCE REQUIREMENTS as in the Standard Technical Specifications.

Relaxing the rod OPERABILITY test surveillance from 14 to 30 days has no impact on control rod availability because of the insignificant number of control rod drive failures determined by the current bi-weekly surveillance test. The proposed surveillance reduction will also have the benefit of decreasing the likelihood of inadvertently dropping a rod during the test and reducing wear on the rod drive mechanism from the surveillance test. The proposed 30 day test interval is also consistent with industry practice in that 30 days is the Standard Technical Specification surveillance interval.

In summary, the proposed relaxations of current Technical Specification requirements do not significantly increase the probability of or consequences of a previously evaluated accident because: The 0.3% reactivity limit is not necessary to preserve any Safety analysis assumption, the hi-flux trip setpoint reduction is not as appropriate a requirement to compensate for an INOPERABLE rod position deviation and flux deviation alarm as the increased surveillance, and the 30 day surveillance to verify rod OPERABILITY, combined with other rod position surveillance requirements will adequately verify rod OPERABILITY.

b. Create the possibility of a new or different kind of accident from any previously analyzed because the proposed change introduces no new mode of plant operation nor involves a physical modification to the plant.

c. Involve a significant reduction in a margin of safety. As discussed in item 3a above, the 0.3% reactivity limit is not a restriction based on any safety analysis assumption, the hi-flux setpoint reduction is not necessary to compensate for any adverse impact of an INOPERABLE rod position and flux deviation alarm in the safety analysis, and the 31 day OPERABLE rod surveillance is consistent with industry practice and the Standard Technical Specifications.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.1.3.1 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: POSITION INDICATING SYSTEM - OPERATING

NO: 3/4.1.3.2

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.2.5, and Table 4.1-1, items 9 and 10.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more complete than the current Technical Specification as follows:

1. A new LCO is added which defines the position indication system OPERABILITY requirement of determining rod position within ± 12 steps.

2. ACTION limits are added which increase the rod position surveillance frequency from once per 12 hours to once per 8 hours if one of the rod position indication systems is inoperable.

3. The MODE applicability is expanded to include MODE 2.

c. The revision relaxes the following current requirement.

Surveillance intervals for rod position indicators are relaxed from 8 to 12 hours.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed changes as described in Item 2.a are similar to example (i) of 48 FR 14870 in that they are administrative changes which consolidate current requirements into a technical specification format consistent with the Standard Technical Specifications and do not involve technical or plant modifications.

2) The proposed changes as described in Items 2.b.1, 2.b.2 and 2.b.3 are similar to example (ii) of 48 FR 14870 in that they provide additional restrictions and controls by including an LCO defining rod position OPERABILITY requirements (± 12 steps), ACTION limits on increased rod position surveillance (12 hours to 8 hours) when a rod position indicator is inoperable and by including MODE 2 in the MODE APPLICABILITY requirement.

3) The proposed change to relax the rod position indication system surveillance from 8 to 12 hours does not involve a significant hazards consideration because this change would not:

    a. Involve a significant increase in the probability of or consequences of an accident previously evaluated.

The rod position indication system surveillance is relaxed from 8 hours to 12 hours. Manual surveillance of control rod position supplements the continuous surveillance of rod position by the rod deviation monitor which actuates an annunciator when any rod exceeds a rod misalignment limit. The proposed requirement includes increasing the periodic surveillance to once per 4 hours whenever the rod deviation monitor is inoperable. Because the rod deviation monitor provides continuous surveillance of rod position and the periodic surveillance is increased if this alarm is inoperable, relaxing the rod surveillance to 12 hours has no impact on any previously evaluated accident.

b. Create the possibility of a new or different kind of accident from any previously analyzed because the proposed change introduces no new mode of plant operation nor involves a physical modification to the plant.

c. Involve a significant reduction in a margin of safety because the rod position indication system surveillance is only relaxed to the interval required by the group step demand position Technical Specification (12 hours). The surveillance interval is also consistent with industry practice in that 12 hours is the rod position indication system surveillance interval allowed by the Standard Technical Specifications.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.1.3.2 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: POSITION INDICATION SYSTEM - SHUTDOWN

NO: 3/4.1.3.3

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

There is no corresponding LCO requirement in the current Turkey Point Technical Specifications. A related Surveillance Requirement is contained in Table 4.1-1, item 9.

2) Proposed Condition of License:

a. The amendment adds a new Technical Specification that specifies the LCO, APPLICABLE MODES, ACTION statements and SURVEILLANCE REQUIREMENTS for the POSITION INDICATION SYSTEM in MODES 3, 4 and 5.

b. The revision is more complete than the current Technical Specification as follows:

1. A new LCO is added which contains the POSITION INDICATION SYSTEM REQUIREMENTS applicable to MODE 3 thru 5.

2. An explicit ACTION statement is included which requires that the Reactor Trip System breakers be opened when the required POSITION INDICATION SYSTEM is inoperable.

3. A SURVEILLANCE REQUIREMENT is added to demonstrate the GROUP DEMAND POSITION INDICATION SYSTEM OPERABILITY

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

Proposed Tech. Spec. No.3/4.1.3.3

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The changes in Item 2.a is similar to example (i) of 48 FR 14870 in that they are administrative changes which consolidate current requirements into a technical specification format consistent with the Standard Technical Specifications and do not involve technical or plant modifications.

2) The proposed changes as described in Items 2.b.1, 2.b.2 and 2.b.3 are similar to example (ii) of 48 FR 14870 in that they provide a new LCO defining DEMAND POSITION INDICATION SYSTEM requirements for MODES 3, 4 and 5, an ACTION statement to open the reactor trip system breakers when the DEMAND POSITION INDICATION SYSTEM is inoperable, and a SURVEILLANCE REQUIREMENT that demonstrates that the DEMAND POSITION INDICATION SYSTEM is OPERABLE.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.1.3.3 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: ROD DROP TIME

NO: 3/4.1.3.4

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.2.3 and Table 4:1-2, item 5.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more restrictive than the current Technical Specification because all control rods are required to meet the rod drop time limit prior to power operation.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed changes as described in Item 2.a are similar to example (i) of 48 FR 14870 in that they are administrative changes which consolidate current requirements into a technical specification format consistent with the Standard Technical Specifications and do not involve technical or plant modifications.

2) The proposed change as described in Item 2.b is similar to example (ii) of 48 FR 14870 in that an additional limitation is included to require all control rod drop times to be equal to or less than the rod drop time limit prior to power operation.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.1.3.4 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: SHUTDOWN ROD INSERTION LIMIT

NO: 3/4.1.3.5

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.2.1.a.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more complete than the current Technical Specification as follows:

1. The MODE APPLICABILITY includes MODE 2.

2. The ACTION statement requires corrective action within 1 hour, when the SHUTDOWN ROD INSERTION LCO is violated.

3. The SURVEILLANCE REQUIREMENTS verify that the SHUTDOWN RODS are fully withdrawn at least once per 12 hours.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed changes as described in Item 2.a are similar to example (i) of 48 FR 14870 in that they are administrative changes which consolidate current requirements into a technical specification format consistent with the Standard Technical Specifications and do not involve technical or plant modifications.

2) The proposed changes as described in Items 2.b.1, 2.b.2 and 2.b.3 are similar to example (ii) of 48 FR 14870 in that they provide additional restrictions and controls by including SHUTDOWN ROD INSERTION LIMITS in MODE 2, an ACTION statement time limit of 1 hour for corrective action and a 12 hour SURVEILLANCE REQUIREMENT of the SHUTDOWN ROD INSERTION LIMIT.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.1.3.5 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: CONTROL ROD INSERTION LIMITS

NO: 3/4.1.3.6

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.2.1.b, c, d, and g.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more complete than the current Technical Specification as follows:

1. The CONTROL ROD INSERTION LIMIT LCO is applied in MODE 2.

2. The ACTION statement requires corrective action within 2 hours if the LCO is violated.

3. The control rod bank position is required to be surveilled at least once per 12 hours.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed changes as described in Item 2.a are similar to example (i) of 48 FR 14870 in that they are administrative changes which consolidate current requirements into a technical specification format consistent with the Standard Technical Specifications and do not involve technical or plant modifications.

2) The proposed changes as described in Items 2.b.1, 2.b.2 and 2.b.3 are similar to example (ii) of 48 FR 14870 in that they provide additional restrictions and controls including CONTROL ROD INSERTION LIMITS in MODE 2, a 2 hour ACTION time limit to restore the LCO condition, and a 12 hour control rod bank position SURVEILLANCE to ensure that the LCO limit is not violated.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.1.3.6 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: AXIAL FLUX DIFFERENCE (AFD)

NO: 3/4.2.1

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

   As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.2.6.c thru 3.2.6g and 3.2.8.

2) Proposed Condition of License:

   a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

   b. The revision is more complete than the current Technical Specification as follows:

      1. Explicit time limits are allowed for corrective action.

      2. The surveillance interval for monitoring the AXIAL FLUX DIFFERENCE is explicitly defined.

   c. The revision relaxes the following current requirement.

      The AFD mode applicability is restricted to MODE 1, above 15% power.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed changes as described in Item 2.a are similar to example (i) of 48 FR 14870 in that they are administrative changes which consolidate current requirements into a technical specification format consistent with the Standard Technical Specifications and do not involve technical or plant modifications.

2) The proposed changes as described in Items 2.b.1 and 2.b.2 are similar to example (ii) of 48 FR 14870 in that they provide added restrictions and controls in the form of explicit time limits (30 minutes with power above 50% and 15 minutes with power above 90%) for corrective action when an LCO violation occurs, and explicit SURVEILLANCE REQUIREMENTS for monitoring the AXIAL FLUX DIFFERENCE (once per 7 days).

3) The proposed change to relax the AXIAL FLUX DIFFERENCE LCO to power levels above 15% does not involve a significant hazards consideration because this change would not:

   a. Involve a significant increase in the probability of or consequences of an accident previously evaluated.

      The relaxed Technical Specification requirement described in 2.c above is justified because the core vendor has determined that at power levels below 15% no significant fission product inventory can be built into the core which could cause a perturbation to the power distribution when the plant returns to high power. Because this relaxation has no impact on any safety analysis assumption involving the core power distribution there is no impact on any previously evaluated accident. Further, the proposed MODE APPLICABILITY requirement is consistent with industry practice in that it is the same as the MODE APPLICABILITY in the Standard Technical Specifications.

b. Create the possibility of a new or different kind of accident from any previously analyzed because the proposed change introduces no new mode of plant operation nor involves a physical modification to the plant.

c. Involve a significant reduction in a margin of safety because as explained above, the proposed relaxation has no impact on the safety analysis.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.2.1 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: HEAT FLUX HOT CHANNEL FACTOR - F_Q (Z)

NO: 3/4.2.2

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.2.6.a. and b and Table 4.1-1, Item 1b.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision relaxes the current requirement to remeasure the HEAT FLUX HOT CHANNEL FACTOR F_Q within 24 hours whenever a measured value of F_Q exceeds its limit. The proposed requirement allows continued operation with reduced power and trip setpoints and requires F_Q to be demonstrated to be within its limit prior to increasing power above the reduced power limit.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature.

1) The proposed changes as described in Item 2.a are similar to example (i) of 48 FR 14870 in that they are administrative changes which consolidate current requirements into a technical specification format consistent with the Standard Technical Specifications and do not involve technical or plant modifications.

2) The proposed change to relax the time interval between F_Q Surveillances when the F_Q limit is violated, does not involve a significant hazards consideration because this change would not:

   a. Involve a significant increase in the probability of or consequences of an accident previously evaluated.

      Both the current and proposed Technical Specification ACTIONS require an immediate reduction in power when an F_Q limit is violated. Because the F_Q limit increases as power decreases this required ACTION is sufficient to restore the F_Q dependent conditions assumed in the safety analysis. The second F_Q surveillance within 24 hours required by the current Technical Specification will most likely only serve to confirm the results of the first F_Q surveillance. Because the proposed Technical Specification ACTION restriction to reduce power adequately preserves the safety analysis conditions that depend on F_Q the proposed relaxation has no impact on the probability of or consequences of any previously evaluated accidents.

      Further, the proposed change is consistent with industry practice in that the same ACTION restrictions are contained in the Standard Technical Specification.

   b. Create the possibility of a new or different kind of accident from any previously analyzed because the proposed change introduces no new mode of plant operation nor involves a physical modification to the plant.

   c. Involve a significant reduction in a margin of safety because the proposed change includes an immediate ACTION to reduce power by 1% for each 1% that the F_Q limit is exceeded. This power reduction is adequate to compensate for any adverse impact of the F_Q limit violation on the safety analysis.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.2.2 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

NO: 3/4.2.3

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.2.6.a and b and Table 4.1-1, Item 1b.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more restrictive than the current Technical Specification as follows:

The ACTION statement requires a power reduction to 50% and the hi-flux trip setpoint reduced to 55% within 2 hours after $F^N_{\Delta H}$ exceeds its limit. In the current requirements, power is reduced in proportion to the $F^N_{\Delta H}$ limit violation.

c. The revision relaxes the following current requirement:

The current requirement to go to HOT STANDBY within 24 hours of an $F^N_{\Delta H}$ limit violation is replaced with a power limit of 5%.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed changes as described in Item 2.a are similar to example (i) of 48 FR 14870 in that they are administrative changes which consolidate current requirements into a technical specification format consistent with the Standard Technical Specifications and do not involve technical or plant modifications.

2) The proposed change as described in Item 2.b is similar to example (ii) of 48 FR 14870 in that a power restriction of 50% is more restrictive than a power reduction that is proportional to the $F^N_{\Delta H}$ limit violation.

3) The proposed change to replace the required MODE reduction (to HOT STANDBY) with a 5% power limit, does not involve a significant hazards consideration because this change would not:

   a. Involve a significant increase in the probability of or consequences of an accident previously evaluated.

   The proposed change does not represent any practical relaxation in requirements because a 5% power limit does not allow practical sustained power operation in MODE 1. The only significant difference between the current requirements and the proposed requirements is that the proposed requirement may avoid a forced MODE reduction if an $F^N_{\Delta H}$ limit violation occurs and cannot be corrected within 24 hours. The ACTION requirement in the proposed change to reduce power to 5% is sufficient to compensate for the adverse impact of any credible $F^N_{\Delta H}$ limit violation and, therefore, the proposed change will not significantly increase the probability of or consequences of any previously evaluated accident.

   Further, the proposed change is consistent with industry practice in that the same ACTION restrictions are contained in the Standard Technical Specifications.

b. Create the possibility of a new or different kind of accident from any previously analyzed because the proposed change introduces no new mode of plant operation nor involves a physical modification to the plant.

c. Involve a significant reduction in a margin of safety because the proposed change includes a power reduction to 5%, which is adequate to compensate for any adverse impact of the $F^N_{\Delta H}$ limit violation on the safety analysis.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.2.3 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: QUADRANT POWER TILT RATIO (QPTR)

NO: 3/4.2.4

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.2.6h and i.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

A reporting requirement in the current Technical Specification to report the QPTR violation as an abnormal occurrence is included as a requirement for a SPECIAL REPORT as defined in Section 6.9.2 of the proposed Technical Specifications with a 30 day reporting requirement.

A current requirement to notify the NRC if core hot channel factors are not determined is deleted.

b. The revision is more restrictive than the current Technical Specification as follows:

1. Larger reductions in power level and trip setpoints are required in the proposed Technical Specification for a given QPTR penalty (3% versus 2% for each % the QPTR exceeds 1.0).

2. If the QPTR is not restored within its limit within 24 hours for a QPTR violation less than 1.09, or within 2 hours if the QPTR violation is greater than 1.09, the core power level and hi-flux trip setpoints must be reduced to 50% and 55% respectively.

3. Surveillance intervals are explicitly defined, including increased surveillance frequency if the associated QPTR alarm is inoperable (once per 12 hours versus once per 7 days). In-core detectors are also required for QPTR surveillance if an ex-core detector is inoperable.

4. The proposed Technical Specification requires power and trip setpoint reductions with an indicated QPTR violation. The current Technical Specification allows the option of re-measuring core peaking factors and basing operating limits with the QPTR violation on the incore measurements rather than the QPTR violation.

c. The revision relaxes the following current requirements:

1. MODE APPLICABILITY is relaxed to MODE-1, POWER above 50%.

2. A current requirement to reduce the overtemperature and overpower delta-T (OTΔT and OPΔT) trip setpoints, in addition to the hi-flux trip setpoint is deleted.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed changes as described in Item 2.a are similar to example (i) of 48 FR 14870 in that they are administrative changes which consolidate current requirements into a technical specification format consistent with the Standard Technical Specifications and do not involve technical or plant modifications.

2) The proposed changes as described in Items 2.b.1 through 2.b.4 are similar to example (ii) of 48 FR 14870 in that the added restrictions on the required power and Trip Setpoint reductions are larger, surveillance requirements are made more restrictive if related monitoring alarms or instruments are inoperable and options for using in-core instrumentation as an alternative to the ex-core nuclear instruments are less flexible. The proposed Technical Specification reduces core power and trip setpoints 3% per % violation, rather than 2% per % violation. If a QPTR violation of 9% or less lasts for 24 hours, or a 9% or greater violation lasts for 2 hours, core power must be further reduced to 50% or less and the hi-flux trip setpoint must be reduced to 55% or less.

   Surveillance frequencies increase with inoperable instruments includes increasing from 7 days to 12 hours the QPTR surveillance when a QPTR alarm is inoperable. With a power range nuclear instrument channel inoperable the proposed Technical Specification requires that the QPTR surveillance be based on data from two sets of four symmetric in-core thimbles or a full core flux map.

   The proposed changes are more restrictive in that core power and trip setpoint reductions are required when a QPTR limit violation occurs regardless of a violation in the power distribution limits.

3) The proposed change to relax the MODE APPLICABILITY requirement to MODE 1 above 50% power and to delete the OTΔT and OPΔT setpoint reductions, does not involve a significant hazards consideration because these changes would not:

   a. Involve a significant increase in the probability of or consequences of an accident previously evaluated.

      The MODE APPLICABILITY relaxation will not significantly increase the probability of or consequences of an accident previously evaluated because the core design organization has determined generically that at 50% power and below QUADRANT POWER TILTS cannot contribute to any significant core thermal penalty.

      The current Technical Specification includes setpoint reductions in the high flux, OT-Delta-T and OP-Delta-T trip setpoints. The proposed Technical Specification includes setpoint reductions in the high flux trip setpoint only. However, when compared to the current Technical Specification requirements for trip point reduction, the proposed Technical Specification requires a large reduction in one setpoint rather than smaller reductions in several setpoints. It is judged that the proposed Technical

Specification results in a negligible safety reduction compared to the current Technical Specification because the setpoint reduction in the hi flux setpoint adequately compensates for the QPTR penalty. The added benefit associated with other setpoint reductions only contributes to overall trip system reliability because of the increased diversity of the trip system when multiple diverse trip signals are included. With the coincidence trip system logic using multiple redundant trip channels each trip function is highly reliable and the benefit resulting from diverse trips is minimal.

In addition, the relaxed requirements described above are consistent with industry practice in that the proposed MODE APPLICABILITY and trip channel setpoint reductions are consistent with the Standard Technical Specifications.

b. Create the possibility of a new or different kind of accident from any previously analyzed because the proposed change introduces no new mode of plant operation nor involves a physical modification to the plant.

c. Involve a significant reduction in a margin of safety because the core design organization has determined generically that at 50% power and below the inherent safety margin gained from the power reduction exceeds any potential safety margin reduction from a QUADRANT POWER TILT.

The deletion of the OTΔT and OPΔT trip setpoint reductions will also not significantly reduce any safety margin because of the required reduction in the hi-flux setpoint, as explained above.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.2.4 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: DNB PARAMETERS

NO: 3/4.2.5

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.1.6.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more restrictive than the current Technical Specification as follows:

1. The indicated values of DNB parameters are listed as limits and include allowances for instrumentation uncertainty.

2. The SURVEILLANCE REQUIREMENT specifies verification of RCS flow at least once per 12 hours.

3. The SURVEILLANCE REQUIREMENT specifies Channel Calibration of RCS flow rate indicators at least once per refueling.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety; then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed changes as described in Item 2.a are similar to example (i) of 48 FR 14870 in that they are administrative changes which consolidate current requirements into a technical specification format consistent with the Standard Technical Specifications and do not involve technical or plant modifications.

2) The proposed changes as described in Items 2.b.1, 2.b.2 and 2.b.3 are similar to example (ii) of 48 FR 14870 in that they provide additional restrictions by including an allowance for instrument uncertainty, a more frequent RCS flow surveillance and a requirement to calibrate the RCS flow indicator.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.2.5 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: REACTOR TRIP SYSTEM INSTRUMENTATION

NO: 3/4.3.1

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.5.1 and Table 4.1-1.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more complete than the current Technical Specification as follows:

1. A complete list of trip channels is included in Table 3.3-1.

2. Trip Channel OPERABILITY requirements are included for MODES 3, 4 and 5.

c. The revision relaxes the following current requirement:

The current bi-weekly surveillance interval for the OTΔT and OPΔT analog channel operational test is relaxed to monthly.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed changes as described in Item 2.a are similar to example (i) of 48 FR 14870 in that they are administrative changes which consolidate current requirements into a technical specification format consistent with the Standard Technical Specifications and do not involve technical or plant modifications.

2) The proposed changes as described in Item 2.b.1 and 2.b.2 are similar to example (ii) of 48 FR 14870 in that added restrictions and controls in the form of a more complete list of trip channels is included in the list of trip channels required to be OPERABLE and trip channel OPERABILITY requirements for MODES 3, 4 and 5 are included.

3) The proposed change to relax the OTΔT and OPΔT trip channel surveillance intervals from bi-weekly to monthly, does not involve a significant hazards consideration because this change would not:

   a. Involve a significant increase in the probability of or consequences of an accident previously evaluated.

      The analog channel operational test verifies that the trip channels are able to perform their trip function. Past experience at Turkey Point over a typical 12 month interval consisting of OTΔT and OPΔT bi-weekly trip functional surveillance tests have shown that trip channels failed the surveillance procedure acceptance criteria on only 5 of 150 tests. Furthermore, the surveillance procedure acceptance criteria is more restrictive than the Technical Specification acceptance criteria for setpoints and includes both high and low side setpoint drift.

      Based on this observed trip channel reliability the proposed relaxed surveillance interval will not degrade the trip system reliability. Therefore this change will not significantly increase the probability of or consequences of any previously evaluated accident.

      In addition, the monthly surveillance of OTΔT and OPΔT trip channel is consistent with industry practice in that it is the surveillance interval in the Standard Technical Specifications.

b. Create the possibility of a new or different kind of accident from any previously analyzed because the proposed change introduces no new mode of plant operation nor involves a physical modification to the plant.

c. Involve a significant reduction in a margin of safety because of the high reliability of the OTΔT and OPΔT trip channels as demonsrated by the current surveillance test results.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.3.1 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: ENGINEERED SAFETY FEATURE ACTUATION (ESFAS)

NO: 3/4.3.2

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.5, Table 3.5-2, 3.5-3, 3.5-4, Table 4.1-1.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more complete than the current Technical Specification as follows:

1. The ESFAS instrumentation Table 3.3-2 defining channel operability and Mode appliability contains a more complete list of ESFAS instrumentation.

2. The minimum channels operable column has been changed in detail such that Table 3.3-2 is more restrictive than the current requirements.

3. The ACTION requirements for inoperable channels have been rewritten and are more restrictive than the current requirements, except as noted below in 2.c.

4. The ESFAS instrument surveillance testing includes a more detailed list of components in the ESFAS channels.

c. The revision relaxes the following current requirements:

1. In the event that one channel of manual initiation of selected ESFAS functions is inoperable, a 48 hour allowed outage time is included for restoration of the channel to OPERABLE status before shutdown is required. For the manual initiation of a steamline isolation, a 48 hour

allowed outage time is included for restoration of the channel to OPERABLE status before following the action required by Specification 3.7.1.5. The current specifications allow only 1 hour before initiating a plant status change.

2. In the ESFAS trip setpoint table, an allowance for channel drift is included in each setpoint allowable value.

BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which consolidates current requirements into a technical specification format consistent with the Standard Technical Specifications and does not involve technical or plant modifications.

2) The proposed changes as described in Items 2.b.1, 2.b.2, 2.b.3 and 2.b.4 are similar to example (ii) of 48 FR 14870 in that they provide additional limitations and controls by including additional instruments in Table 3.3-2, requiring a greater number of channels and more restrictive ACTION statements for same instrument channels, and expanded instrument component details in the surveillance tables.

3) The proposed change to relax the action time to recover an inoperable manual ESFAS initiation channel from one hour to 48 hours does not involve a significant hazards consideration because this change would not:

   a. Involve a significant increase in the probability of or consequences of an accident previously evaluated. Manual initiation of ESFAS by the operator is considered a backup to the automatic trip functions. One channel of manual initiation remains OPERABLE during the time allowed to recover the out of service channel.

      The proposed revision is consistent with industry practice in that the 48 hour outage time is the same as that required in the Standard Technical Specifications.

   b. Create the possibility of a new or different kind of accident from any previously analyzed because the proposed change introduces no new mode of plant operation nor involve a physical modification to the plant.

   c. Involve a significant reduction in a margin of safety because the operator has one channel of manual initiation available during the allowed ACTION statement time.

4) The proposed change to relax the action time to recover an inoperable manual steamline isolation channel from one hour to 48 hours does not involve a significant hazards consideration because this change would not:

   a. Involve a significant increase in the probability of or consequences of an accident previously evaluated. Manual initiation of Steam Line Isolation by the operator is considered a backup to the automatic trip functions.

      The specification is consistent with industry practice in that the 48 hour outage time is the same as that required in the Standard Technical Specifications .

   b. Create the possibility of a new or different kind of accident from any previously analyzed because the proposed change introduces no new mode of plant operation nor involve a physical modification to the plant.

c. Involve a significant reduction in a margin of safety because the automatic methods of steamline isolation initiation are available during the allowed ACTION statement time.

5) The proposed change to relax the Instrument Setpoint Table by including an Allowable Value column does not involve a significant hazards consideration because this change would not:

a. Involve a significant increase in the probability of or consequences of an accident previously evaluated. The setpoint allowable value is to define an amount of change that can be attributed to drift and not cause the channel to be inoperable. The setpoint allowances are consistent with safety analysis assumptions hence do not constitute an extension of the plant operating envelope.

b. Create the possibility of a new or different kind of accident from any previously analyzed because the proposed change introduces no new mode of plant operation nor involve a physical modification to the plant.

c. Involve a significant reduction in a margin of safety because the required calibration setpoints are the same as the current technical specification and the allowed drift is consistent with the safety analyses.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.3.2 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: RADIATION MONITORING FOR PLANT OPERATION

NO: 3/4.3.3.1

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification Table 3.5-3 Item 4, Table 3.5-4 Item 10, Table 4.1-1 Item 18A, 18B, 36 and 38.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more complete than the current Technical Specification as follows:

1. An ACTION statement requiring that the alarm Setpoints meet the tabulated limits has been added.

2. The containment radioactivity monitors are required to be operable in all modes.

3. The spent fuel radioactivity monitors have been added and are required to be operable whenever there is irradiated fuel in the spent fuel pits.

4. The control room air intake radiation monitors are added.

5. The CHANNEL CHECK of containment radiation monitors is required to be performed once per shift instead of daily.

6. Calibration of containment radioactivity monitors is added to SURVEILLANCE REQUIREMENTS.

c. The revision relaxes the following requirement:

The SURVEILLANCE REQUIREMENT of area radiation monitors is deleted.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which consolidates current requirements into a technical specification format consistent with the Standard Technical Specifications and does not involve technical or plant modifications.

2) The proposed changes as described in Items 2.b.1, 2.b.2, 2.b.3, 2.b.4 and 2.b.5 are similar to example (ii) of 48 FR 14870 in that they provide additional limitations, restrictions and controls by including an ACTION statement requiring alarm/setpoints meet tabulated limits or a channel is to be declared inoperable and applicability for the containment radioactivity monitor has been extended to all modes. Both spent fuel and control room air intake radioactivity monitors have been added to the specification. Surveillance for CHANNEL CHECK on containment radiation monitors has been increased from daily to once per shift.

3) The proposed change to relax the requirement to include the Area Radiation Monitor in the Technical Specification does not involve a significant hazards consideration because this change would not:

a. Involve a significant increase in the probability of or consequences of an accident previously evaluated. The Area Radiation Monitoring System is designed to monitor radiation levels at various locations within the operating area of the two units and provide an early warning of a potential unsafe health condition that way have developed. The area radiation monitor provides no automatic function for protection of reactor or plant systems during postulated accidents. Other radiation monitors that are included in the revised technical specifications are designed for and would provides indications that a malfunction has occured that has resulted in increased radiation levels in the containment building, reactor coolant, or other process systems.

b. Create the possibility of a new or different kind of accident from any previously analyzed because the proposed change introduces no new mode of plant operation nor involve a physical modification to the plant.

c. Involve a significant reduction in a margin of safety because the radiation monitors do not perform any automatic accident mitigating function. The area radiation monitors will be maintained OPERABLE in accordance with plant procedure and are backed up by area radiation surveys. The specification is consistent with industry practice in that area radiation monitors are not included in the Standard Technical Specification.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.3.3.1 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: MOVABLE INCORE DETECTOR

NO: 3/4.3.3.2

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.2.7.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more complete than the current Technical Specification as follows:

1. The total number of OPERABLE incore detector thimbles has been increased from 32% to 75%.

2. The APPLICABILITY is now defined as during monitoring of QPTR and measurement of $F^{N}_{\Delta H}$ and $FQ^{(Z)}$.

3. The ACTION statement does not allow continued operation of the reactor if the Incore Detector System doesn't meet its operability requirement. The current specification allows reduced power level operation without the Incore Detectors meeting the operability conditions.

4. The new SURVEILLANCE requires that Incore Detection System be demonstrated operable at least once per 24 hours when required to make neutron flux measurements.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which consolidates current requirements into a technical specification format consistent with the Standard Technical Specifications and does not involve technical or plant modifications.

2) The proposed changes as described in Items 2.b.1, 2.b.2, 2.b.3 and 2.b.4 are similar to example (ii) of 48 FR 14870 in that they provide additional limitation by including an increase in the required number of operable detector thimbles and definition of the Applicability for the specification. Deletion of reduced power operation from the ACTION statement and adding the new surveillance requirements provide additional restrictions and controls on the Movable Incore Detectors.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.3.3.2 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: SEISMIC INSTRUMENTATION

NO: 3/4.3.3.3

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification Table 4.1-1 Item 25.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more complete than the current Technical Specification as follows:

1. Appropriately worded LCO, APPLICABILITY, and ACTION statements have been added.

2. A surveillance requiring the seismograph film to be replaced with fresh film at least once per 12 months has been added.

3. A surveillance requiring CHANNEL CALIBRATION to be performed at least once per 18 months has been added.

4. A special surveillance has been added including a report submittal, and CHANNEL CALIBRATION if the seismic monitoring instrument experiences a seismic event.

5. The current requirement of battery replacement semi-annually is deleted because a battery charger has been installed and if during quarterly battery test, the battery is found to be weak or does not meet vendor's specification, the battery will be replaced. The charger was recently added and significantly improves the availability of the battery when required.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which consolidates current requirements into a technical specification format consistent with the Standard Technical Specifications and does not involve technical or plant modifications.

2) The proposed changes as described in Items 2.b.1, 2.b.2, 2.b.3, 2.b.4 and 2.b.5 are similar to example (ii) of 48 FR 14870 in that they provide additional restrictions and controls by including the LCO, APPLICABILITY and ACTION statements using the requirements of the Standard Technical Specifications. Additional controls were added by requiring annual replacement of the film, 18 month CHANNEL CALIBRATION, rewording to reflect the new battery chargers and new special reporting requirements and calibrations if a seismic event occurs.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.3.3.3 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: METEOROLOGICAL INSTRUMENTATION

NO: 3/4.3.3.4

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

The current Turkey Point Unit 3 and 4 Technical Specification does not explicitly specify requirements for METEOROLOGICAL INSTRUMENTATION.

2) Proposed Condition of License:

a. The revision is more complete than the current Technical Specification as follows:

This new Technical Specification is being proposed to be added. Addition of this specification will assure that proper METEOROLOGICAL information is available when needed to analyze the effects of plant operations.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed change as described in Item 2.a is similar to example (ii) of 48 FR 14870 in that it provides a new Technical Specification with stated limitations, restrictions and controls for the METEOROLOGICAL INSTRUMENTS applicable at all times.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.3.3.4 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: ACCIDENT MONITORING INSTRUMENTATION

NO: 3/4.3.3.5

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Table 3.5-5, Table 4.1-1 Items 6, 17A, 17B, 24, 26, 27, 28, 29, 30, 34, 35, 36, 37, 38 and 39, Table 4.18-1 Item 12, and the Proposed Licensing Amendment transmitted to the NRC by L-86-296 dated July 18, 1986.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more complete than the current Technical Specification as follows:

1. The following instruments have been added:

a) Reactor Coolant Pressure (wide range)
b) Reactor Hot Leg Temperature (wide range)
c) Reactor Cold Leg Temperature (wide range)
d) Reactor Vessel Level Monitoring System (License Amendment Request L-86-296, dated July 18, 1986)

2. The submittal for special report is now required in 14 days rather than 30 days if either the Containment High Range Radiation Monitor or the High Range Noble Gas Monitor is INOPERABLE for more than 7 days.

3. The monthly functional test of Containment High Range Area Radiation Monitor is deleted because the CHANNEL CHECK test performed every shift is the functional test. The CHANNEL CHECK definition specifies that Area Radiation Monitors be exposed to a source. The Radiation Monitors when exposed to a source actuate the alarms and other interlocks thus fulfilling the functional test. The deletion of functional test requirement is considered administrative in nature as CHANNEL CHECK is retained which meets the functional test requirement.

c. The revision relaxes the following current requirement:

1. Function Test and Calibration required for Emergency Portable Survey Instruments is deleted.

2. Monthly flowpath verificaion of the Post Accident Sampling System is deleted.

3. Boric Acid Tank and Refueling Water Storage Tank level instruments have a relaxed CHANNEL CHECK from weekly to monthly intervals.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed changes as described in Items 2.a and 2.b.3 are similar to example (i) of 48 FR 14870 in that they are administrative changes which consolidate current requirements into a technical specification format consistent with the Standard Technical Specifications and do not involve technical or plant modifications and remove duplicate test requirements from a tabular listing.

2) The proposed changes as described in Items 2.b.1 and 2.b.2 are similar to example (ii) of 48 FR 14870 in that they provide

additional limitations and controls by including the Reactor Coolant Pressure, Reactor Coolant Hot Leg Temperature, Reactor Coolant Cold Leg Temperature and Reactor Vessel Level Monitoring System in the tabular listings. Additional controls are included in the revision by specifying a shorter time before a special report must be submitted when either of the Containment High range Radiation or Noble Gas Monitor Channels are INOPERABLE for longer than 7 days.

3) The proposed change to relax the requirement for Functional Test and Calibration of Emergency Portable Survey instruments does not involve a significant hazards consideration because this change would not:

a. Involve a significant increase in the probability of or consequences of an accident previously evaluated. The deletion of Calibration of Emergency Portable Survey Instruments from the current Technical Specifications is made because it is understood that when portable instrumentation is used it should have the same level of surveillance as required for instrumentation it replaces. The routine calibration of portable instrumentation is covered in plant procedures and its deletion from the proposed Technical Specifications is consistent with industry practice in that it is not required by Standard Technical Specification.

b. Create the possibility of a new or different kind of accident from any previously analyzed because the proposed change introduces no new mode of plant operation nor involve a physical modification to the plant.

c. Involve a significant reduction in a margin of safety because the portable instruments are not used for plant automatic protection functions and the calibration program for these instruments is not changed.

4. The proposed change to relax the requirement for verification of the PASS Flow Paths does not involve a significant hazards consideration because this change would not:

a. Involve a significant increase in the probability of or consequences of an accident previously evaluated. The requirement of PASS flowpath verification is deleted as it is covered in proposed Technical Specification 6.8 that specifies a program be established for PASS. The PASS program ensures the capability to obtain and analyze the reactor coolant, radioactive iodines and particulates in the plant gaseous effluents, and containment atmosphere samples under accident conditions. The program also includes training of personnel, procedures for sampling and analysis, and provisions for maintenance of sampling and analysis equipment. The proposed changes are consistent with industry practice and the Standard Technical Specifications.

b. Create the possibility of a new or different kind of accident from any previously anlyzed because the proposed change introduces no new mode of plant operation nor involve a physical modification to the plant.

c. Involve a significant reduction in a margin of safety because the PASS System functionality is assured by the program and procedural requirements of Specification 6.8.

5. The No Significant Hazards Evaluation for the relaxed CHANNEL CHECK frequency for the Boric Acid Tank and Refueling Water Storage Tank level instruments is presented with the Boration Water Sources - Operating Specification 3/4.1.2.6.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.3.3.5 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTS

NO: 3/4.3.3.6

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.9.1.c, Tables 3.9-2 and 4.1-3.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more complete than the current Technical Specification as follows:

The length of time radioactive releases are permitted with inoperable monitoring instrumentation has been reduced from 30 days to 14 days under administrative control and is more clearly stated.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which consolidates current requirements into a technical specification format consistent with the Standard Technical Specifications and does not involve technical or plant modifications.

2) The proposed change as described in Item 2.b is similar to example (ii) of 48 FR 14870 in that it provides a more restrictive limitation by reducing the time radioactive releases may continue with inoperative monitoring instrumentation.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.3.3.6 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

NO: 3/4.3.3.7

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.9.2d, Tables 3.9-4 and 4.1-4.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more complete than the current Technical Specification as follows:

1. During inoperability of Gas Decay Tank Explosive Gas Monitoring System, the revision requires grab samples be taken once per 4 hours. The current Technical Specification requires grab samples be taken once per 8 hours.

2. The revision allows condenser air ejector releases for 14 days with inoperable monitoring instrumentation provided grab samples are taken. The current Technical Specification allows release up to 30 days.

3. During condenser air ejector flow rate and sampler flow rate monitoring instrumentation inoperability, the revision requires that flow estimation be performed once per 4 hours. The current Technical Specification requires flow estimation once per 8 hours.

4. During Plant vent or spent fuel vent sampler flow rate monitoring instrumentation inoperability, the revision requires that flow estimation be performed once per 4 hours. The current Technical Specification requires flow estimation once per 8 hours.

5. The revision requires that gas decay tank explosive gas monitoring instrumentation be calibrated using one volume percent hydrogen; four volume percent hydrogen; one volume percent oxygen; and four volume percent oxygen (all balance nitrogen). The current Technical Specification requires calibration using one volume percent oxygen; and four volume percent oxygen (balance nitrogen).

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which consolidates current requirements into a technical specification format consistent with the Standard Technical Specifications and does not involve technical or plant modifications.

2) The proposed changes as described in Items 2.b.1, 2.b.2, 2.b.3, 2.b.4 and 2.b.5 are similar to example (ii) of 48 FR 14870 in that they provide more restrictions and limitations by reducing the time between grab samples from 8 to 4 hours for gas decay tank, air ejector flow rate estimates and plant vent samples flow rate estimates. An additional limitation is also included by reducing the time condenser air ejector releases can continue from 30 to 14 days when monitoring instrumentation is inoperable. Additional controls are included by requiring two additional concentrations of hydrogen or oxygen when calibrating the waste gas decay tank explosive gas monitor.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.3.3.7 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: TURBINE OVERSPEED PROTECTION

NO: 3/4.3.4

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification Table 4.1-2 Items 15 and 16.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the SURVEILLANCE REQUIREMENTS.

b. The revision is more complete than the current Technical Specification as follows:

1. LCO is added that explicitly states the requirement for operability of overspeed protection system.

2. APPLICABILITY modes are explicitly stated for the LCO.

3. ACTION statement is added which imposes a turbine shutdown if Turbine Control/Stop Valves do not meet operability tests.

4. SURVEILLANCE REQUIREMENTS require:

a. Demonstration of operability of turbine valves once per 31 days by cycling each valve through at least once complete cycle from running position. The current requirement specifies check of valve closure on a monthly basis and allows a 50% tolerance on time between tests. The revision allows a 25% tolerance.

b. That at least once per 31 days, movement of each valve through one complete cycle verified by direct observation be performed. The current Technical Specification does not specify this.

c. CHANNEL CALIBRATION of overspeed protection system once per 18 months. The current Technical Specification does not specify this.

d. disassembly of at least one valve, once per 40 months, and performance of visual and surface inspection. The SURVEILLANCE REQUIREMENT specifies complete inspection of the valves such as: seats, disks, stems and corrosion. The detail inspection of the valves and a requirement to inspect all valves of a given type if a problem is found provides assurance that the overspeed protection will stop steam flow to the turbine.

c. The revision relaxes the following current requirement:

The requirement for visual, magnetic particle and dye penetrant inspection of the Low Pressure Turbine Rotor is deleted.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which consolidates current requirements into a technical specification format consistent with the Standard Technical Specifications and does not involve technical or plant modifications.

2) The proposed changes as described in Items 2.b.1, 2.b.2, 2.b.3 and 2.b.4 are similar to example (ii) of 48 FR 14870 in that they provide additional limitation by including an explicit LCO requiring operability of the TURBINE OVERSPEED PROTECTION SYSTEM and the required modes. An ACTION statement imposes additional restriction requiring a shutdown of the turbine or closure of the nonfunctional flowpath when a portion of the system is inoperable. Additional controls are included by the reduction in the time tolerance between surveillance tests of the turbine valves and by requiring a full cycle of the valves being tested. Additional controls are included by requiring a CHANNEL CALIBRATION once per refueling and a valve disassembly and detail inspection of one sample of each valve type every 40 months.

3. The proposed change to relax the requirement for a visual, magnetic particle and dye penetrant inspection of the Low Pressure Turbine Rotor for Technical Specification requirements does not involve a significant hazards consideration because this change would not:

   a. Involve a significant increase in the probability of or consequences of an accident previously evaluated. The requirement to perform NDE on the Low Pressure Turbine Rotor is to provide early warning of fatigue in the rotor metal thus reducing the possibility of missiles from a rotor failure. In addition procedural controls on load increase rates, metal temperature changes, differential expansion between the rotor and casing, rotor eccentricity and vibration measured at the bearings provide inservice monitoring of the turbine to insure that stress levels are kept acceptably low. Many of the above parameters are charted on a recorder thus making trends easy to detect.

   b. Create the possibility of a new or different kind of accident from any previously analyzed because the proposed change introduces no new mode of plant operation nor involves a physical modification to the plant.

   c. Involve a significant reduction in a margin of safety because any serious degradation of the turbine rotating elements would cause a change in the monitored turbine parameters or be observed during the vendor's inspection program. In-service inspection of the turbine ensures that flaws arising during turbine operation are detected and repaired long before they become even a potential challenge to turbine structural integrity. FPL complies with the turbine vendor's NRC approved inspection schedule and refurbishment recommendations for the Turkey Point Plant turbines.

Elimination of this SURVEILLANCE REQUIREMENT is consistent with industry practice in that it is not included in the Standard Technical Specification.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.3.4 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: REACTOR COOLANT LOOPS AND COOLANT CIRCULATION - STARTUP AND POWER OPERATION

NO: 3/4.4.1.1

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

   As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.1.1.a.1, 3.1.1.a.3, 3.1.1.a.4, 3.4.1.c and Table 4.1-2, Item 18.

2) Proposed Condition of License:

   a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

   b. The revision is more restrictive than the current Technical Specification as follows:

      The revised LCO is applicable in more MODES (MODE 2).

   c. The revision relaxes the following current requirement.

      The allowed outage time for a REACTOR COOLANT LOOP in MODE 1 is relaxed from one hour to six hours.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed changes as described in Item 2.a are similar to example (i) of 48 FR 14870 in that they are administrative changes which consolidate current requirements into a technical specification format consistent with the Standard Technical Specifications and do not involve technical or plant modifications.

2) The proposed change as described in Item 2.b is similar to example (ii) of 48 FR 14870 in that the added MODE 2 applicability requirements represent an additional restriction on plant operation.

3) The proposed change to relax the allowed outage time for a REACTOR COOLANT LOOP from one to six hours, does not involve a significant hazards consideration because this change would not:

   a. Involve a significant increase in the probability of or consequences of an accident previously evaluated.

      Relaxing the time limit to be in HOT STANDBY from one to six hours will allow the plant additional time to restore the loop or perform a normal shutdown. Increasing this ACTION statement time limit will have a minimal impact on a previously evaluated accident because the ACTION statement only applies in the unlikely event of a single RCS loop being lost during MODE 1 or 2. With power above the P-8 interlock setpoint of 45% the loss of a loop will result in an automatic reactor trip. With power below the P-8 setpoint, a second plant accident transient during the time interval of the ACTION statement is unlikely. The Reactor Trip System continues to monitor plant conditions during the ACTION time interval and trip functions such as overtemperature delta-T, or loss of flow are available to provide protection during the ACTION time interval. Finally, adopting the proposed ACTION time has the potential benefit of reducing the number of reactor trip transients imposed on the plant.

The proposed allowed outage time limit of six hours is consistent with industry practice in that it is the allowed outage time limit in the Standard Technical Specifications.

These considerations demonstrated that the proposed six hour allowed outage time limit will not significantly increase the probability of or consequences of any previously evaluated accident.

b. Create the possibility of a new or different kind of accident from any previously analyzed because the proposed change introduces no new mode of plant operation nor involves a physical modification to the plant.

c. Involve a significant reduction in a margin of safety because of the extremely unlikely combination of events that are required to occur before the allowed outage time relaxation can impact the plant safety margin. These events include: the loss of one RCS pump while the remaining two pumps continue to operate, a core power level below the P-8 interlock setpoint (45%) and a second accident transient that occurs within the six hour allowed outage time which is not mitigated by the reactor trip system.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.4.1.1 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: REACTOR COOLANT LOOPS AND COOLANT CIRCULATION - HOT STANDBY

NO: 3/4.4.1.2

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.1.1.a.2, 3.4-1.d and Table 4.1-2, Item 18.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more restrictive than the current Technical Specification as follows:

More coolant loops are required to be operating in a shutdown MODE if the scram breakers are closed.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which consolidates current requirements into a technical specification format consistent with the Standard Technical Specifications and does not involve technical or plant modifications.

2) The proposed change as described in Item 2.b is similar to example (ii) of 48 FR 14870 in that it provides added restrictions on the required number of operating RCS loops in MODE 3 if the reactor trip system breakers are closed.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.4.1.2 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: REACTOR COOLANT LOOPS AND COOLANT CIRCULATION - HOT SHUTDOWN

NO: 3/4.4.1.3

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

   As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 1.23, 3.1.1.a.2, 3.1.1.a.5, 3.4.1.e and Table 4.1-2, Item 18.

2) Proposed Condition of License:

   a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

   b. The revision is more restrictive than the current Technical Specification as follows:

      Restrictions are placed on starting a Reactor Coolant Pump when potentially large temperature differences exist between the primary coolant and steam generator secondary water.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout

the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed changes as described in Item 2.a are similar to example (i) of 48 FR 14870 in that they are administrative changes which consolidate current requirements into a technical specification format consistent with the Standard Technical Specifications and do not involve technical or plant modifications.

2) The proposed change as described in Item 2.b is similar to example (ii) of 48 FR 14870 in that added restrictions and controls are placed on the secondary to primary steam generator temperature difference prior to starting a Reactor Coolant Pump (<50 F).

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.4.1.3 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: REACTOR COOLANT LOOPS AND COOLANT CIRCULATION - COLD SHUTDOWN - LOOPS FILLED

NO: 3/4.4.1.4.1

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.1.1.a.2, 3.1.1.a.5, 3.4.1.e and Table 4.1-2, Item 18.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more complete than the current Technical Specification as follows:

The LCO provided applies to MODE-5 with the RCS loops filled. In this specification the operating COOLANT LOOP must be an RHR loop. With the RCS loops not full, Specification 3/4.4.1.4.2 requires that the two OPERABLE COOLANT LOOPS must be RHR loops. Current requirements allow an RCS or RHR loop to be the required coolant loop and do not address an unfilled RCS loop condition.

c. The revision relaxes the following current requirements:

The current Technical Specification which requires an OPERABLE REACTOR COOLANT PUMP and a steam generator secondary water level of 10% or more when an RCS loop is used for decay heat dissipation is replaced with the requirement that the RCS loop steam generator secondary water level be above 10% in two steam generators.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed changes as described in Item 2.a are similar to example (i) of 48 FR 14870 in that they are administrative changes which consolidate current requirements into a technical specification format consistent with the Standard Technical Specifications and do not involve technical or plant modifications.

2) The proposed change as described in Item 2.b is similar to example (ii) of 48 FR 14870 in that the more restrictive requirement that the operating coolant loop be an RHR loop and that both OPERABLE coolant loops be RHR loops if the RCS loops are not full is included.

3) The proposed change to relax the RCS loop OPERABLE requirement by replacing it with a steam generator level requirement, does not involve a significant hazards consideration because this change would not:

   a. Involve a significant increase in the probability of or consequences of an accident previously evaluated.

The Standard Technical Specification allows use of a steam generator for backup residual heat removal. It has been shown generically by the core designer that with some inventory in the secondary side of the steam generator natural circulation will develop and a steam generator with the prescribed secondary water level can be used to dissipate decay heat in place of an RHR loop. However, the proposed Technical Specification only uses the steam generator as a backup heat sink to the required operating RHR loop.

Using steam generator instead of an RCS loop as the backup heat sink to the operating RHR loop has no impact on any previously evaluated accident because the steam generator can effectively dissipate decay heat without the RCS pump running. In addition, the proposed Technical Specification ACTION requirement, in the event that the RHR loop is lost, is to take immediate corrective ACTION to restore the RHR loop.

b. Create the possibility of a new or different kind of accident from any previously analyzed because the proposed change introduces no new mode of plant operation nor involves a physical modification to the plant.

c. Involve a significant reduction in a margin of safety because the above relaxation has been determined to have no impact on any previously evaluated accident.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.4.1.4.1 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: REACTOR COOLANT LOOPS AND COOLANT CIRCULATION - COLD SHUTDOWN - LOOP NOT FILLED

NO: 3/4.4.1.4.2

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.1.1.a.2, 3.1.1.a.5, 3.4.1.e and Table 4.1-2, Item 18.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more complete than the current Technical Specification as follows:

The LCO provided applies to MODE-5 with the RCS loops not filled. The operating COOLANT LOOP must be an RHR loop. Current requirements allow an RCS or RHR loop to be the required coolant loop and do not address an unfilled RCS loop condition.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed changes as described in Item 2.a are similar to example (i) of 48 FR 14870 in that they are administrative changes which consolidate current requirements into a technical specification format consistent with the Standard Technical Specifications and do not involve technical or plant modifications.

2) The proposed change as described in Item 2.b is similar to example (ii) of 48 FR 14870 in that the more restrictive requirement that the operating coolant loop be an RHR loop and that both OPERABLE coolant loops be RHR loops if the RCS loops are not full is included.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.4.1.4.2 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: SAFETY VALVES - SHUTDOWN

NO: 3/4.4.2.1

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.1.1.c.1, Table 4.1-2 Item 6 and B3.1.1.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more complete than the current Technical Specification as follows:

1. The LCO has the SAFETY VALVE setpoint and tolerance added.

2. The applicable modes have been defined more clearly.

3. An ACTION statement has been added which is more appropriate for the mode of plant operation covered by the specification.

4. The proposed revision references the SAFETY VALVE testing to the requirement of the ASME Section XI in Specification 4.0.5.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which consolidates current requirements into a technical specification format consistent with the Standard Technical Specifications and does not involve technical or plant modifications.

2) The proposed changes as described in Items 2.b.1, 2.b.2, 2.b.3 and 2.b.4 are similar to example (ii) of 48 FR 14870 in that they provide additional limitation and control by adding the required Safety Valve Setpoint to the LCO and the applicable modes are more clearly defined. Also included is an ACTION statement appropriately worded for the SHUTDOWN mode and the reference of the SURVEILLANCE REQUIREMENT to Specification 4.0.5.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.4.2.1 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: SAFETY VALVES - OPERATING

NO: 3/4.4.2.2

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.1.1.c.2, Table 4.1-2 Item 6 and B3.1.1.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more complete than the current Technical Specification as follows:

1. The LCO has the SAFETY VALVE setpoint and tolerance added.

2. An ACTION statement has been reduced the time allowed to make an inoperable valve operable from 1 hour to 15 minutes.

3. The proposed revision references Specification 4.05 for SAFETY VALVE testing.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which consolidates current requirements into a technical specification format consistent with the Standard Technical Specifications and does not involve technical or plant modifications.

2) The proposed changes as described in Items 2.b.1, 2.b.2, and 2.b.3 are similar to example (ii) of 48 FR 14870 in that they provide additional limitation and controls by adding the required Safety Valve Setpoint to the LCO, revising the ACTION statement to decrease the time to restore an inoperable safety valve and the reference of Specification 4.0.5 for the SURVEILLANCE REQUIREMENTS.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.4.2.2 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: PRESSURIZER

NO: 3/4.4.3

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.1.1.d and Table 4.2-1 Item 2.1 thru 2.8.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more complete than the current Technical Specification as follows:

1. The LCO has included a maximum water level and increased the number of heater groups from one to two.

2. An ACTION statement has been added.

3. The following surveillance requirements have been added:

a) A surveillance requiring 12 hour checks of PRESSURIZER level,

b) A surveillance to measure heater group input power every 92 days and

c) A surveillance that verifies the pressurizer emergency power source once per refueling.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which consolidates current requirements into a technical specification format consistent with the Standard Technical Specifications and does not involve technical or plant modifications.

2) The proposed changes as described in Items 2.b.1, 2.b.2, and 2.b.3 are similar to example (ii) of 48 FR 14870 in that they provide additional limitation, restrictions and controls by including PRESSURIZER level limit to the LCO, ACTION statement requiring shutdown when the PRESSURIZER is inoperable and adding surveillances for PRESSURIZER level, heater input power and emergency power sources.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.4.3 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: PORV BLOCK VALVES

NO: 3/4.4.4

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.1.1.e.1, 3.1.1.e.2 and 3.1.1.e.3.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more complete than the current Technical Specification as follows:

The revised Technical Specification requires that the PORV Block Valve be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle and observing valve position indication. The current Technical Specifiction does not specify valve cycling.

c. The revision relaxes the following requirement:

The PORV's are not required to be operable.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which consolidates current requirements into a technical specification format consistent with the Standard Technical Specifications and does not involve technical or plant modifications.

2) The proposed change as described in Item 2.b is similar to example (ii) of 48 FR 14870 in that it provides additional restriction by requiring the PORV block valve be demonstrated operable by cycling the valve and verifying the position indication every 92 days.

3) The proposed change to relax the requirement to demonstrate OPERABILITY of the PORV's in Specification 3/4.4.4 for MODES 1, 2 and 3 does not involve a significant hazards consideration because the change would not:

   a. Involve a significant increase in the probability of or consequences of an accident previously evaluated. Reactor Coolant System overpressure protection is provided by the Pressurizer Safety Valves as addressed in Specification 3/4.4.2.

      The Steam Generator tube rupture accident does require a means to depressurize the Reactor Coolant System to reduce coolant leakage to the Secondary Side of the Steam Generator. The primary means of depressurizing the primary System is by use of the normal pressurizer spray. Auxiliary pressurizer sprays can be used as a backup. While the PORV can be used as a second backup it is the least desirable because it tends to reduce Reactor Coolant System inventory.

   b. Create the possibility of a new or different kind of accident from any previously analyzed because the proposed change introduces no new mode of plant operation nor involves a physical modification to the plant.

c. Involve a significant reduction in a margin of safety because the accident analyses does not rely on the PORV's as a primary means for accident mitigation. However if the PORV's were required it is very unlikely that the PORV would not be availale as a backup because the surveillances on operability of PORV(s) are required by specifications 3/4.4.9.3 (Overpressure Protection System) and 3/4.3.3.5 (Accident Monitoring Instrumentation).

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.4.4 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: STEAM GENERATORS

NO: 3/4.4.5

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 4.2.5.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more complete than the current Technical Specification as follows:

1. An ACTION statement has been added.

2. The inspection criteria and SURVEILLANCE REQUIREMENT tables have been converted to match the Inservice Inspection Program.

c. The requirement for entry from the hot leg side of the steam generator to the first support of the cold leg has been modified to allow concurrent inspection from both hot and cold legs.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which consolidates current requirements into a technical specification format consistent with the Standard Technical Specifications and does not involve technical or plant modifications.

2) The proposed changes as described in Items 2.b.1 and 2.b.2 are similar to example (ii) of 48 FR 14870 in that they provide additional restriction and controls by adding an ACTION statement and by more clearly stating the STEAM GENERATOR inspection criteria and SURVEILLANCE REQUIREMENTS.

3) The proposed change to relax the current requirement identified in 2c does not involve a significant hazards consideration because this change will not :

   a. Involve a significant increase in the probability of or consequences of an accident previously evaluated.

      The eddy current testing techniques in use make no distinction in as to the point of entry. Wherever practical, all tests are run from tube sheet to tube sheet regardless of the point of entry. Therefore, routinely conducted examinations will be capable of detecting problems either on the hot leg or cold leg side.

      The safety aspect of the steam generator eddy current inspection is to allow the detection, and removal from service, of any tube with indications of a defect greater than 40% through wall. In view of the fact that the eddy current exam is equally effective regardless of the point of entry, and that for the vast majority of tubes the examination is from tube sheet to tube sheet, it is highly unlikely that an indication will be missed by this change Therefore, this is no significant increase in the probability of, or consequences of an accident previously evaluated.

b. Create the possibility of a new or different kind of accident because the proposed change introduces no new mode of plant operation nor involves a physical modification to the plant.

c. Involve a significant margin of safety. The reliability of the inspection technique remains unchanged, as does the required plugging limit. Therefore, the required margin of safety remains unchanged.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.4.5 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: LEAKAGE DETECTION SYSTEMS

NO: 3/4.4.6.1

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.1.3f and B3.1.3.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more complete than the current Technical Specification as follows:

1. The number of OPERABLE LEAKAGE DETECTION SYSTEMS required by the LCO has been increased from two to three.

2. The applicability of the LCO for LEAKAGE DETECTION SYSTEMS has been expanded from greater than 2 percent power to MODES 1, 2 and 3.

3. The proposed ACTION statement is more specific and appropriately worded for the three detection systems. The proposed revision requires eventual shutdown of the plant but provides alternative methods for monitoring while awaiting repair. More emphasis is given to operability of the radiation monitoring systems as they are more sensitive to detection of small leaks than the sump level system.

4. Specific SURVEILLANCE REQUIREMENTS have been added for both gaseous and particulate radiation detection as well as the sump level system.

c. The revision relaxes the requirement to allow the radioactive monitoring system to be inoperable from 48 hours to 7 days.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which consolidates current requirements into a technical specification format consistent with the Standard Technical Specifications and does not involve technical or plant modifications.

2) The proposed changes as described in Items 2.b.1, 2.b.2, 2.b.3 and 2.b.4 are similar to example (ii) of 48 FR 14870 in that they provide additional limitations by increasing the required number of LEAKAGE DETECTION SYSTEMS from two to three. Additional restrictions have been imposed by expanding the applicability from greater than 2 percent power to MODES 1, 2 and 3 and expanding the ACTION statement to cover three detection systems. Additional controls have been added by including specific SURVEILLANCE REQUIREMENTS for all three of the LEAKAGE DETECTION SYSTEMS.

3) The proposed change to relax the current requirement identified in 2c does not involve a significant hazards consideration because this change will not:

   a. Involve a significant increase in the probability of or consequences of an accident previously evaluated.

The design of the Turkey Point Units 3 and 4 radioactive particulate and gaseous air monitors is such that samples are drawn from a single containment penetration through common valves and equipment. The failure of a common component such as the pump unit or valve would cause both the particulate and gaseous detection systems to be inoperable until repairs were made.

The current technical specification allows 48 hours to troubleshoot the problem, make repairs, recalibrate the system and place it back into operation. Plant experience has indicated that the 48 hour repair time is not sufficient.

The proposed revision would allow both the gaseous and particulate systems to be inoperable for 7 days. This change would not cause a significant increase in the probability of or consequence of an accident previously evaluated because the revised technical specification would require that during the period the instruments are inoperable the following explicit requirements be implemented:

1) The reactor cavity sump level monitoring system be operable,

2) Grab samples be obtained and analyzed at least once per 24 hours, and

3) A reactor coolant system water level inventory balance be performed at least once per 8 hours during steady state operation, except when operating in the shutdown cooling mode.

The proposed change is consistant with industry practice in that other plants have similar technical specification requirements.

b. Create the possibility of a new or different kind of accident because the proposed change introduces no new mode of plant operation nor involves a physical modification to the plant.

c. Involve a significant margin of safety. As discussed in 3a above the proposed revision requires that specific actions be taken to monitor potential leakage while the radioactive detection systems are inoperable. These actions are not specified in the current technical specifications. These new actions would increase the margin of safety during the time period that the radioactive detection systems were inoperable.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.4.6.1 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: OPERATIONAL LEAKAGE

NO: 3/4.4.6.2

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.1.3a, b, c, d, e and g; 3.16, 4.17 and Table 4.1-2 Item 11.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more complete than the current Technical Specification as follows:

1. The proposed revision requires that with any pressure boundary leakage, the unit be placed in HOT STANDBY within 6 hours and in COLD SHUTDOWN within following 30 hours. The current Technical Specification states that if system boundary cannot be isolated the reactor be shutdown and cooldown initiated within 24 hours.

2. The proposed revision requires that any leakage greater than the stated LCO limits, excluding Pressure Boundary Leakage, be reduced to within the LCO limit within 4 hours or the unit be shut down.

   The current Technical Specification requires that any leakage be investigated and evaluation initiated within 4 hours, (except for isolation valves) and if the leakage is proven real, reactor SHUTDOWN be initiated within 24 hours. For isolation valves the current Technical Specification requires restoration to within the limit in 6 hours prior to SHUTDOWN action.

3. The proposed revision requires that the Leakage Detection Systems be monitored once per 12 hours. The current specification requires daily evaluation.

4. The proposed revision includes both Residual Heat Removal System Pump suction valves (MOV-750 and MOV-751) in the Pressure Isolation Valve list consistent with IST submittal letter L-85-204.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which consolidates current requirements into a technical specification format consistent with the Standard Technical Specifications and does not involve technical or plant modifications.

2) The proposed changes as described in Items 2.b.1, 2.b.2, 2.b.3 and 2.b.4 are similar to example (ii) of 48 FR 14870 in that they provide additional restrictions and controls by including shorter times before initiating a plant shutdown if leakage exceeds the established limits, additional and more restrictive surveillance requirements and a more frequent evaluation of plant leakage.

Proposed Tech. Spec. No.3/4.4.6.2

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.4.6.2 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: REACTOR COOLANT SYSTEM - CHEMISTRY

NO: 3/4.4.7

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.1.5 and Table 4.1-2 Item 1b.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more complete than the current Technical Specification as follows:

1. The proposed revision restates the ACTION statement to separate the 24 hours of corrective action into mode related statements.

2. The proposed revision requires that at other times (other than Modes 1, 2, 3 and 4) if the steady state limit is exceeded for more than 24 hours or the transient limit is exceeded, pressurizer pressure be reduced to less than or equal to 500 psig, if applicable, and an engineering evaluation be performed. The current Technical Specification requires that the unit be placed in cold shutdown and corrective action be taken.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which consolidates current requirements into a technical specification format consistent with the Standard Technical Specifications and does not involve technical or plant modifications.

2) The proposed changes as described in Items 2.b.1 and 2.b.2 are similar to example (ii) of 48 FR 14870 in that they provide additional restrictions by including mode related ACTION statement and requiring reduction of plant operating pressure and an engineering evaluation if CHEMISTRY LIMITS are exceeded in MODES other than 1, 2, 3 or 4.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.4.7 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: REACTOR COOLANT SYSTEM - SPECIFIC ACTIVITY

NO: 3/4.4.8

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

   As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.1.4, B3.1.4 and Table 4.1-2 Item 1.

2) Proposed Condition of License:

   a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

   b. The revision is more complete than the current Technical Specification as follows:

      1. The proposed revision specifically states the modes in which sample and analysis is required.

      2. The proposed revision requires that Gross radioactivity be determined at least once per 72 hours. The current requirement is for Gross Beta, and Gamma determination and allows 3 days time between samples. The revision is more restrictive as specification 4.0.2 allows less grace period consistent with the Standard Technical Specifications.

      3. The revision moves the current requirements for accumulative time and reporting requirements to Specification 6.9.1. These changes are consistent with Generic Letter No. 85-19, 9/27/85.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement. Example (vii) relates to a change to make a license conform to changes in regulations where the license change result in very minor changes to facility opration clearly in keeping with regulations.

1) The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which consolidates current requirements into a technical specification format consistent with the Standard Technical Specifications and does not involve technical or plant modifications.

2) The proposed changes as described in Items 2.b.1 and 2.b.2 are similar to example (ii) of 48 FR 14870 in that they provide additional restriction by including the specific modes for sample analysis every 72 hours.

3) The proposed change as described in Item 2.b.3 is similar to example (vii) of 48 FR 14870 in that it provides for a change to conform with Generic Letter Number 85-19, 9/27/85.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.4.8 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: PRESSURE/TEMPERATURE LIMITS - REACTOR COOLANT SYSTEM

NO: 3/4.4.9.1

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.1.2, B3.1.2, 4.20 and B4.20.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more complete than the current Technical Specification as follows:

1. The ACTION statement requires that if the LCO limit is exceeded, an engineering evaluation be performed to determine effects on RCS structural integrity.

2. The revision adds a requirement that during heatup, cooldown or pressure testing the RCS temperature and pressure be determined to be within the limits once per 30 minutes.

3. The revision clearly states the surveillance requirement that reactor vessel material specimens be removed and examined to determine changes in material properties as specified by 10CFR50, Appendix H, in accordance with the schedule in Table 4.4-4.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

Proposed Tech. Spec. No. 3/4.4.9.1

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which consolidates current requirements into a technical specification format consistent with the Standard Technical Specifications and does not involve technical or plant modifications.

2) The proposed changes as described in Items 2.b.1, 2.b.2 and 2.b.3 are similar to example (ii) of 48 FR 14870 in that they provide additional restriction and controls by including the requirement for engineering evaluation if LCO limits are exceeded, and added surveillance requirement to determine RCS temperature and pressure compliance every 30 minutes during plant status changes plus a reactor vessel material examination schedule.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.4.9.1 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: PRESSURE/TEMPERATURE LIMITS - PRESSURIZER

NO: 3/4.4.9.2

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.1.2 and B3.1.2.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more complete than the current Technical Specification as follows:

1. The ACTION statement requires that if the LCO limit is exceeded, and engineering evaluation be performed to determine effect on structural integrity of the PRESSURIZER.

2. The revision adds a requirement that the PRESSURIZER TEMPERATURES be verified to be within the limits at least once per 30 minutes during heatup or cooldown.

3. The revision adds a requirement that the spray water temperature differential be determined to be within the limit at least once per 12 hours during auxiliary spray operation.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which consolidates current requirements into a technical specification format consistent with the Standard Technical Specifications and does not involve technical or plant modifications.

2) The proposed changes as described in Items 2.b.1, 2.b.2 and 2.b.3 are similar to example (ii) of 48 FR 14870 in that they provide additional restrictions and controls by including the requirement for engineering evaluation if LCO limits are exceeded, an added surveillance requirement to verify pressurizer temperature every 30 minutes during pressurizer heatup or cooldown and to determine a required temperature differential every 12 hours if pressurizer auxiliary spray is in operation.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.4.9.2 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: OVERPRESSURE PROTECTION SYSTEM

NO: 3/4.4.9.3

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.15, 4.16, B3.15 and B4.15.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

Footnotes are added to clarify required plant operations such as valve maintenance, accumulator fill or flow tests.

b. The revision is more complete than the current Technical Specification as follows:

1. The revision requires that with one PORV inoperable, restore the inoperable PORV to operable status within 7 days or depressurize the RCS through the vent within 8 hours. The 8 hour time limit for depressurization has been added.

2. The revision requires that with both PORVs inoperable, depressurization and Venting of the RCS within 8 hours as the only action to be taken. The current Technical Specification allows 24 hours to restore an OPERABLE PORV before depressurization must be achieved.

3. The revision requires that surveillance of the open PORV Isolation Valve be performed more frequently by reducing the interval from weekly to every 72 hours.

4. The revision requires the RCS vent(s) shall be verified to be open at least once per 12 hours when the vent(s) is being used for overpressure protection.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which consolidates current requirements into a technical specification format consistent with the Standard Technical Specifications and does not involve technical or plant modifications.

2) The proposed changes as described in Items 2.b.1, 2.b.2, 2.b.3 and 2.b.4 are similar to example (ii) of 48 FR 14870 in that they provide additional restrictions and controls by reducing the time interval in the ACTION statements to depressurize the RCS and more frequent surveillance on the PORV's and RCS vent valves when used for overpressure protection.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.4.9.3 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: REACTOR COOLANT SYSTEM - STRUCTURAL INTEGRITY

NO: 3/4.4.10

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specifications 4.2 and 4.3, and Proposed License Amendments transmitted by FPL Letters L-85-346 dated October 11, 1985 and L-86-73 dated February 21, 1986.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more complete than the current Technical Specification as follows:

The revision incorporates LCO and ACTION statements regarding structural integrity of ASME Code Class 1, 2, and 3 components. The current Technical Specification only addresses RCS pressure boundary components

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which consolidates current requirements into a technical specification format consistent with the Standard Technical Specifications and does not involve technical or plant modifications.

2) The proposed change as described in Item 2.b is similar to example (ii) of 48 FR 14870 in that it provides additional limitation by specifying ASME 1, 2 and 3 components rather than just the Reactor Coolant System. The Specification is consistent with industry practice in that the intent of the Specification is the same as that required in the Standard Technical Specification.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.4.10 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: REACTOR COOLANT SYSTEM VENTS

NO: 3/4.4.11

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.1.1.f, 4.19, B3.1.1 and B4.19.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more complete than the current Technical Specification as follows:

The revision adds an additional surveillance that requires RCS vent path downstream valve be demonstrated operable once per 92 days by operating the valve through one complete cycle.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

Proposed Tech. Spec. No. 3/4.4.11

1) The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which consolidates current requirements into a technical specification format consistent with the Standard Technical Specifications and does not involve technical or plant modifications.

2) The proposed change as described in Item 2.b is similar to example (ii) of 48 FR 14870 in that it provides additional control by including the quarterly testing of the downstream valves.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.4.11 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: ACCUMULATORS

NO: 3/4.5.1

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.4.1.a.3, 3.4.1.b.1, 4.5.2.b.3, Table 4.1-1 Item 21 and Table 4.1-2 Item 10.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more restrictive than the current Technical Specification as follows:

1. The LCO applicability include MODE 3.

2. An upper bound on boron concentration of 2350 ppm is included.

3. An upper bound on the ACCUMULATOR nitrogen cover pressure of 675 psig is included.

4. Analog channel operational tests on the level and pressure channels are included.

5. A surveillance to verify that the isolation valves are open once per 12 hours is included.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which consolidates current requirements into a technical specification format consistent with the Standard Technical Specifications and does not involve technical or plant modifications.

2) The proposed changes as described in Item 2.b.1 through 2.b.5 are similar to example (ii) of 48 FR 14870 in that they provide additional restrictions in the form of MODE applicability requirements, upper bounds on the accumulator boron concentration and pressure, and more restrictive surveillances on isolation valve position and on the pressure and level instruments.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.5.1 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: EMERGENCY CORE COOLING SUBSYSTEMS - ECCS SYSTEMS

NO: 3/4.5.2

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.4.1.a.4 thru a.7, 3.4.1.b.2, 3.4.1.b.4 thru b.7, 4.5.1, 4.5.2.a, 4.5.2.b.1 and b.2, 4.5.2.b.4, and Table 4.18-1, Items 1 and 2.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more restrictive than the current Technical Specification as follows:

1. The applicability is more restrictive than the current Technical Specification because it includes MODE 3.

2. The revision requires verification that the ECCS piping is full of water, each valve is in its correct position, and the containment is free of loose debris.

3. In the event that the outage time limit allowed by the ACTION statement is exceeded the revised Technical Specification requires a MODE reduction within 12 hours versus 48 hours allowed by the current Technical Specifications for the Mode reduction.

4. The revision requires verification of interlocks which prevent inadvertent pressurization of the RWST from the RHR System.

c. The revision relaxes the following current requirements:

1. The ACTION statement requirement that inoperable equipment be returned to OPERABLE status within 24 hours has been relaxed to 72 hours.

2. The requirement to cycle the Boron Injection Tank Outlet Valves, Containment Sump Recirculation Valves and RWST Outlet Valves has been relaxed from once every 30 days to the period consistent with the requirements of the inservice inspection programs provided in Technical Specification 4.0.5.

3. The requirement to test the Safety Injection and RHR pumps has been relaxed from once every 30 days to the period consistent with the requirements of the inservice inspection program provided in Technical Specification 4.0.5.

4. The current requirement to go to COLD SHUTDOWN if the LCO is not restored within the ACTION time limit is replaced with the requirement to go to HOT SHUTDOWN.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed changes as described in Item 2.a are similar to example (i) of 48 FR 14870 in that they are administrative changes which consolidate current requirements into a technical specification format consistent with the Standard Technical Specifications and do not involve technical or plant modifications.

2) The proposed changes as described in Items 2.b.1 thru 2.b.4 are similar to example (ii) of 48 FR 14870 in that they provide additional restrictions and controls by the added mode applicability (MODE 3), the more complete flow path surveillance which ensures proper valve lineup and the absence of voids in the flow path, and containment sump surveillance to ensure the absence of loose debris. The time allowed for a MODE reduction if the ACTION statement allowed outage time limit is exceeded is reduced from 48 hours to 12 hours. In addition, the SURVEILLANCE REQUIREMENTS require verification of interlocks which prevent inadvertent pressurization of the RWST from the RHR system.

3) The proposed changes to relax the ACTION statement allowed outage time limit, the pump and valve OPERABILITY surveillance test interval, and the MODE reduction for an LCO violation, does not involve a significant hazard consideration because these changes do not:

   a. Involve a significant increase in the probability of or consequences of an accident previously evaluated.

      The ACTION statement requirement that inoperable equipment be returned to OPERABLE status within 72 hours is consistent with industry practice in that it is the Standard Technical Specification requirement. The probability that a second equipment failure in the redundant ECCS equipment and a Loss of Coolant or Steam Line Rupture Accident that requires the ECCS for accident mitigation might occur within the allowed outage time limit is extremely remote. Therefore, the proposed 72 hour allowed outage time limit will not significantly increase the probability of or consequences of an accident previously evaluated.

      The current Technical Specifications require a Mode reduction to COLD SHUTDOWN if the allowed outage time limit is violated, while in MODE 1. The revised Technical Specifications require all ECCS components to be OPERABLE in MODES 1 thru 3 and a reduced subset of ECCS components to be OPERABLE in MODE 4 (HOT SHUTDOWN). The basis for the reduced ECCS component OPERABILITY requirements in MODE 4 is the reduced probability of a LOCA or Steam Line Rupture accident in MODE 4 and because of the reduced severity of either one of these accidents if initiated from MODE 4.

The revised requirements, by including a MODE 4 ECCS LCO are more flexible than the current requirements which can force a Mode reduction to COLD SHUTDOWN. Because the RHR pump and heat exchanger are common to the RHR and ECCS Subsystems, a forced Mode reduction to COLD SHUTDOWN may require using the RHR System when parts of it are inoperable. Therefore, the proposed requirements for MODE reductions are more flexible than current requirements and may avoid forcing RHR System operation with an inoperable RHR component. Because the proposed requirements include reduced ECCS requirements applicable to MODE 4 and are more flexible than current requirements they do not significantly increase the probability of or consequences of any previously evaluated accident. They are also consistent with industry practice in that they are the Standard Technical Specification requirements.

The requirement to cycle the Boron Injection Tank (BIT) Outlet Valve, Containment Sump Recirculation Valves and the RWST Outlet Valves has been shifted from this Technical Specification to Technical Specification 4.0.5 which requires testing consistent with the inservice test program. The cycling frequency has been relaxed to:

BIT Outlet Valves	-	Every 3 months
Sump Recirculation Valves	-	Every COLD SHUTDOWN
RWST Outlet Valves	-	Every Refueling Outage

Cycling the BIT Outlet Valves every 3 months is acceptable based on the standby status of the system and the Plant's prior experience with the more frequent testing internal which has demonstrated the high reliability of these valves.

The cycling of the Sump Recirculation Valves has been relaxed to every COLD SHUTDOWN consistent with the inservice testing program. Failure of these valves during testing could result in loss of containment integrity and potential loss of the recirculation mode of safety injection.

The cycling of the RWST Outlet Valves has been relaxed to each refueling outage consistent with the inservice testing program, because the failure of either of these valves in the non-open position by testing during plant operation would result in a total loss of system function for the associated Containment Spray System and Low Pressure Safety Injection System.

In addition the failure of the RWST Outlet Valves in the non-open position, by testing during COLD SHUTDOWN, could jeopardize the ability of the associated High Pressure Safety Injection pumps to support a LOCA on the operating unit. These valves are required by Plant Technical Specifications to be open and the breakers locked out during plant operation.

The proposed requirement relaxes the safety injection and RHR pump surveillance from monthly to the requirements of the inservice test program which is based on Section XI of the ASME Code (quarterly). This relaxation is justified based on the high reliability of the pumps as demonstrated by the insignificant number of pump failures detected by the current monthly surveillance tests. In addition, the relaxed surveillance frequency reduces the probability of a system failure caused by human error which can be introduced at the time of the surveillance test itself and reduces wear on the affected components.

In summary, the proposed surveillance intervals will not significantly increase the probability of or consequences of a previously evaluated accident because of the demonstrated high reliability of the pumps and valves based on the Plant's prior test experience. The proposed surveillance also reduces component wear and the probability of a human error during the test that reduces system availability. The proposed surveillance is also consistent with industry practice in that it is the requirement in the Standard Technical Specifications.

b. Create the possibility of a new or different kind of accident from any previously analyzed because the proposed change introduces no new mode of plant operation nor involves a physical modification to the plant.

c. Involve a significant reduction in a margin of safety because the proposed allowed outage time limits, and surveillance intervals will continue to ensure the OPERABILITY of the ECCS System and the MODE reduction will allow a more flexible plant response which reduces dependence on RHR components that may be inoperable.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.5.2 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: ECCS SUBSYSTEM - TAVG LESS THAN 350 F

NO: 3/4.5.3

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

There is no corresponding LCO requirement in the current Turkey Point Technical Specifications.

2) Proposed Condition of License:

a. The amendment adds a new Technical Specification that specifies the LCO, APPLICABLE MODES, ACTION statements and SURVEILLANCE REQUIREMENTS for ECCS SUBSYSTEMS in MODE 4.

The revision is more complete than the current requirements as follows:

1. A new LCO is added which contains ECCS SUBSYSTEMS requirements applicable to MODE 4.

2. An explicit ACTION statement is included which requires restoration of the RWST flow path within 1 hour or go to COLD SHUTDOWN. With an inoperable RHR component, the RCS temperature must be maintained by an alternate heat removal method.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The changes in Items 2.a.1 and 2.a.2 are similar to example (ii) of 48 FR 14870 in that they provide a new LCO for controlling ECCS SUBSYSTEMS in MODE 4 and ACTION statements requiring restoration of the RWST flow path, or use alternate heat removal method, if an RHR component is inoperable.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.5.3 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: REFUELING WATER STORAGE TANK

NO: 3/4.5.4

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.4.1.a.1 and Table 4.1-2 Item 2.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more complete than the current Technical Specification as follows:

1. The revision requires that the RWST water temperature be maintained between 39F and 100F. The current specification does not include this requirement.

2. The revision requires applicability in MODES 1, 2, 3 and 4. The current Technical Specification requires applicability when the reactor is critical.

3. The revision requires that RWST water temperature be verified once per 24 hours whenever the outside air temperature is less than 39 F or greater than 100 F.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed changes as described in Item 2.a are similar to example (i) of 48 FR 14870 in that they are administrative changes which consolidate current requirements into a technical specification format consistent with the Standard Technical Specifications and do not involve technical or plant modifications.

2) The proposed changes as described in Items 2.b.1 thru 2.b.3 are similar to example (ii) of 48 FR 14870 in that they provide additional restrictions and controls by including RWST temperature limits, a temperature surveillance when the ambient temperature exceeds the RWST temperature limits and a more restrictive MODE applicability requirement.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.5.4 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: CONTAINMENT INTEGRITY

NO: 3/4.6.1.1

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.3.1.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more complete than the current Technical Specification as follows:

1) A monthly surveillance for outside containment and 92 days for inside containment has been added to verify that all non-automatic valves, blind flanges and deactivated automatic valves are in the closed position.

2) A monthly surveillance that verifies the containment air lock is operable has been added.

3) A surveillance requiring retest of type B penetrations after each closing to verify that maximum allowable leakage rates have not been exceeded has been added.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The changes in Item 2.a are similar to example (i) of 48 FR 14870 in that the are administrative changes which consolidate current requirements into a technical specification format consistent with the Standard Technical Specifications and do not involve technical or plant modifications.

2) The changes in items 2.b.1, 2.b.2 and 2.b.3 are similar to example (ii) of 48 FR 14870 in that they provide additional controls by added surveillance requirement of periodic valve or penetration seal condition verification, OPERABILITY compliance verification for the air lock and verification that containment penetration leak rates are maintained within specified limits.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.6.1.1 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: CONTAINMENT LEAKAGE

NO: 3/4.6.1.2

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 4.4.1, 4.4.2 and 4.4.3.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more complete than the current Technical Specification as follows:

1. The LCO specifies an overall integrated leakage limit for P_t, the reduced pressure test value.

2. ACTION statement has been added.

3. LCO APPLICABILITY is for MODES 1, 2, 3, and 4. The current Technical Specification does not specifiy MODE APPLICABILITY.

4. The SURVEILLANCE REQUIREMENT concerning Type A testing is added.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The changes in Item 2.a are similar to example (i) of 48 FR 14870 in that they are administrative changes which consolidate current requirements into a technical specification format consistent with the Standard Technical Specifications and do not involve technical or plant modifications.

2) The changes in Item 2.b.2 and 2.b.4 are similar to example (ii) of 48 FR 14870 in that they provide additional restrictions and controls by adding ACTION statements and a SURVEILLANCE REQUIREMENT concerning Type A Testing. The changes in Item 2.b.1 and 2.b.3 are similar to example (ii) of 48 FR14870 in that they provide additional information by including a leakage limit and the applicable plant operating MODES.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.6.1.2 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: CONTAINMENT AIR LOCKS

NO: 3/4.6.1.3

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.3.4 and 4.4.2.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more complete than the current Technical Specification as follows:

1. LCO require that air lock shall be operable with an overall air lock leakage rate of less than or equal to 0.2 L_a at 50 psig. The current Technical Specification does not specifically state leakage rate limit for air lock.

2. The surveillance specifies that provisions of specification 4.0.2 are not applicable for periodic air lock leak rate tests. The current Technical Specification does not have this requirement.

3. The surveillance requirement of verifying that only one door can be opened at a time in each air lock is added.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The changes in Item 2.a are similar to example (i) of 48 FR 14870 in that they are administrative changes which consolidate current requirements into a technical specification format consistent with the Standard Technical Specifications and do not involve technical or plant modifications.

2) The changes in Item 2.b.1, 2.b.2 and 2.b.3 are similar to example (ii) of 48 FR 14870 in that they provide additional information on allowable leakage rate for the airlock, additional restriction on testing intervals and additional surveillance on airlock door interlocks.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.6.1.3 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: CONTAINMENT - INTERNAL PRESSURE

NO: 3/4.6.1.4

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.3.2.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more complete than the current Technical Specification as follows:

The revision is more complete than the current requirement as a surveillance requirement has been added that requires verification of containment internal pressure once per 12 hours.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The change in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which consolidates current requirements into a technical specification format consistent with the Standard Technical Specifications and does not involve technical or plant modifications.

2) The change in Item 2.b is similar to example (ii) of 48 FR 14870 in that it provides additional control by specification of the frequency that containment pressure should be verified within Technical Specification limits.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.6.1.4 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: CONTAINMENT - AIR TEMPERATURE

NO: 3/4.6.1.5

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification does not specify requirements for CONTAINMENT AIR TEMPERATURE.

2) Proposed Condition of License:

a. The revision is more complete than the current Technical Specification as follows:

The amendment adds requirements for containment air temperature including LCO, APPLICABILITY MODES, ACTION statement, and SURVEILLANCE REQUIREMENTS.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The changes in Item 2.a are similar to example (ii) of 48 FR 14870 in that they provide requirements to monitor a plant parameter not included in the previous Technical Specifications. Additional controls have been provided by including a new surveillance for monitoring containment average temperature.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.6.1.5 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: CONTAINMENT VESSEL STRUCTURAL INTEGRITY

NO: 3/4.6.1.6

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

   As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 4.4.5, 4.4.6 and 4.4.7.

2) Proposed Condition of License:

   a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

   b. The revision is more complete than the current Technical Specification as follows:

      1. Limiting Condition for Operation is stated.
      2. ACTION statements are specified.
      3. A specific acceptance criteria of 90% predicted lift off force on a tendon is stated.
      4. The number of tendons sampled is increased from 9 to 10.
      5. A specific value for minimum tendon material tensile strength is stated.
      6. A specific tolerance on tendon retention strength is stated.
      7. A specific value for lift off minimum tensile stress is stated.
      8. Specific inspection criteria for sheathing filler grease quality is stated.
      9. The surveillance of tendon and tendon anchorage and liner plate surfaces required when the plant was new have been reinstated.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident

previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The changes in Item 2.a are similar to example (i) of 48 FR 14870 in that they are administrative changes which consolidate current requirements into a technical specification format consistent with the Standard Technical Specifications and do not involve technical or plant modifications.

2) The changes in Item 2.b.1, 2.b.2, 2.b.3, 2.b.4, 2.b.5, 2.b.6, 2.b.7, 2.b.8, and 2.b.9 are similar to example (ii) of 48 FR 14870 in that they provide additional information or requirements relating to the containment vessel structural integrity.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.6.1.6 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: CONTAINMENT VENTILATION SYSTEM

NO: 3/4.6.1.7

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 4.4.2.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more complete than the current Technical Specification as follows:

1. A Limiting Condition for Operation that requires OPERABILITY of the Containment purge and exhaust valves is stated.

2. APPLICABILITY in MODES 1, 2, 3 and 4 is required.

3. ACTION statements are added per plant design.

4. Surveillance requirements are added that require verification of valve position, and determination of cumulative time the valves have been open per calendar year.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The changes in Item 2.a are similar to example (i) of 48 FR 14870 in that they are administrative changes which consolidate current requirements into a technical specification format consistent with the Standard Technical Specifications and do not involve technical or plant modifications.

2) The changes in Item 2.b.1, 2.b.2, 2.b.3, and 2.b.4 are similar to example (ii) of 48 FR 14870 in that they provide additional controls and requirements by including a statement for the Limiting Condition for Operation, MODE APPLICABILITY, ACTION Statements and specific surveillances on the Containment Purge and Exhaust Valves.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.6.1.7 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: CONTAINMENT SPRAY SYSTEM

NO: 3/4.6.2.1

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.4.2, 4.6 and Table 4.18 Item 4.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more complete than the current Technical Specification as follows:

The revision requires that if inoperable equipment is not restored within the time specified in the applicable LCO, the unit shall be placed in COLD SHUTDOWN within 36 hours. The current Technical Specification requires 48 hours.

c. The revision relaxes the following current requirement:

1) The current Technical Specification allows one CONTAINMENT SPRAY SYSTEM INOPERABLE for up to 24 hours in MODE 1. The proposed revision allows one CONTAINMENT SPRAY SYSTEM INOPERABLE for up to 72 hours, in MODE 1.

2) The requirement to test the Containment Spray Pumps has been relaxed from once every 30 days to the period consistent with the requirements of the inservice inspection program provided in Technical Specification 4.0.5 (quarterly).

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed changes as described in Item 2.a are similar to example (i) of 48 FR 14870 in that they are administrative changes which consolidate current requirements into a technical specification format consistent with the Standard Technical Specifications and do not involve technical or plant modifications.

2) The proposed change as described in Item 2b is similar to example (ii) of 48 FR 14870 in that it provides an additional restriction by decreasing the time allowed to place the plant in COLD SHUTDOWN when operating with an inoperable CONTAINMENT SPRAY SYSTEM train.

3) The proposed changes to relax the out-of-service time requirement and the pump OPERABILITY surveillance test interval do not involve a significant hazards consideration because these changes would not:

   a. Involve a significant increase in the probability of or consequences of an accident previously evaluated. The CONTAINMENT SPRAY SYSTEM provides post-accident cooling of the containment atmosphere. The proposed revision would allow an

increase in the allowed out-of-service time for one of the two containment spray trains from 24 hours to 72 hours. This change would not have a significant impact on consequences of a previously evaluated accident because of the following considerations. The Containment Cooling System is redundant to the Spray System. The probability of loss of both the Containment Cooling Systems is extremely remote. This event would require, during the 48 hour period, the loss of offsite power, the loss of diesel generator powering the OPERABLE Sprays System and the loss of the Containment Cooling System coincident with the need for Containment Spray System.

An additional consideration is that the proposed revisions allowing 72 hours is more restrictive than industry practice in that 7 days is allowed for an inoperable spray pump in the Standard Technical Specifications.

The proposed revision relaxes the Containment Spray Pump surveillance from monthly to the requirements of the inservice test program which is based on Section XI of the ASME Code (quarterly). This relaxation is justified based on the high reliability of the pumps as demonstrated by the insignificant number of pump failures detected by the current monthly surveillance tests. In addition, the relaxed surveillance frequency reduces the probability of a system failure caused by human error which can be introduced at the time of the surveillance test and reduces wear on the affected components. The proposed revision is consistent with industry practice in that it is the required surveillance interval in the Standard Technical Specifications.

b. Create the possibility of a new or different kind of accident from any previously analyzed because the proposed change introduces no new mode of plant operation nor involves a physical modification to the plant.

c. Involve a significant reduction in a margin of safety because of the remote probability that containment cooling would not be provided as discussed in Item a above. In addition the revised 72 hour out-of-service time would adequately allow time for potential repairs and, therefore, would not place the plant in a shutdown transient condition and subsequent startup.

The proposed revision in surveillance intervals will not involve a significant reduction in a margin of safety because of the demonstrated high reliability of the Containment Spray Pumps based on the Plant's prior test experience. The proposed surveillance also reduces component wear and the potential for human error during current more frequent testing.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.6.2.1 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: CONTAINMENT COOLING SYSTEM

NO: 3/4.6.2.2

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specifications 3.4.2, 4.6, and Table 4.18 Item 9.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more complete than the current Technical Specification as follows:

The revision requires that if inoperable equipment is not restored within the time specified in the applicable LCO the Unit shall be placed in COLD SHUTDOWN within 36 hours. The current Technical Specification requires 48 hours.

c. The revision relaxes the following current requirement:

The current Technical Specification allows one CONTAINMENT COOLING unit INOPERABLE for up to 24 hours in MODE 1. The proposed revision allows one CONTAINMENT COOLING unit INOPERABLE for up to 72 hours, in MODE 1.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed changes as described in Item 2.a are similar to example (i) of 48 FR 14870 in that they are administrative changes which consolidate current requirements into a technical specification format consistent with the Standard Technical Specifications and does not involve technical or plant modifications.

2) The proposed change as described in Item 2b is similar to example (ii) of 48 FR 14870 in that it provides an additional restriction by decreasing the time allowed to place the plant in COLD SHUTDOWN when operating with an inoperable CONTAINMENT COOLING UNIT.

3) The proposed change to relax the out-of-service time requirement for one CONTAINMENT COOLING UNIT, does not involve a significant hazards consideration because this change would not:

   a. Involve a significant increase in the probability of or consequences of an accident previously evaluated. The CONTAINMENT COOLING SYSTEM provides post-accident cooling of the containment atmosphere. The proposed revision would allow an increase in the allowed out-of-service time for one of the three Containment Cooling Units from 24 hours to 72 hours. This change would not have a significant impact on consequences of a previously evaluated accident because of the following considerations. The Containment Spray System is redundant to the CONTAINMENT COOLING SYSTEM. The probability of the loss of both the Containment Spray and the Containment Cooling Systems is extremely remote. This event would require, during the 48 hour period, the loss of diesel generator powering the OPERABLE CONTAINMENT COOLING UNITS and the loss of the Containment Spray System coincident with the need for CONTAINMENT COOLING SYSTEM.

   An additional consideration is that the Standard Technical Specification is less restrictive in that it allows 7 days for an inoperable Containment Cooling Unit.

b. Create the possibility of a new or different kind of accident from any previously analyzed because the proposed change introduces no new mode of plant operation nor involvoes a physical modification to the plant.

c. Involve a significant reduction in a margin of safety because of the remote probability that containment cooling would not be provided as discussed in item a above. In addition the revised 72 hour out-of-service time would adequately allow time for potential repairs and, therefore, would not place the plant in a shutdown transient condition and subsequent startup.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.6.2.2 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: EMERGENCY CONTAINMENT FILTERING SYSTEM

NO: 3/4.6.3

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specifications 3.4.3a and 4.7.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more complete than the current Technical Specification as follows:

1. The LCO applicability is required in MODE 1, 2, 3 and 4. The current Technical Specification specifies applicability when reactor is critical.

2. Laboratory analysis of the System's filters is required within 31 days after removal. The current Technical Specification allows 45 days.

3. Verification of filter cooling spray solenoid valves opening by operator action and automatically on a loss of system air flow is required once per refueling. The current Technical Specification does not specify this requirement.

4. The bases section has additional explanation.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which consolidates current requirements into a technical specification format consistent with the Standard Technical Specifications and does not involve technical or plant modifications.

2) The proposed changes as described in Items 2.b.1, 2.b.2, 2.b.3 and 2.b.4 are similar to example (ii) of 48 FR 14870 in that they provide additional restrictions by: specifying applicable MODES of operation; requiring laboratory analysis of the filters within 31 days after removal from the system; verifying that filter cooling spray solenoid valves open by operator action and automatically on a loss of air flow; and adding more information to the bases.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.6.3 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: CONTAINMENT SYSTEMS - CONTAINMENT ISOLATION VALVES

NO: 3/4.6.4

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specifications 3.3.3 and B3.3.3.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more complete than the current Technical Specification as follows:

1. The LCO requires OPERABILITY of additional containment isolation valves. The current Technical Specification specifies OPERABILITY of only Phase "A", "B", and CONTAINMENT VENTILATION ISOLATION VALVES.

2. The revision adds surveillance that requires demonstration of OPERABILITY of valves prior to returning the valve to service after maintenance.

3. The revision adds surveillance that requires demonstration of valve OPERABILITY on actuation of Phase "A", "B", or CONTAINMENT VENTILATION ISOLATION test signal.

4. The revision adds a Table that lists CONTAINMENT ISOLATION VALVES.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which consolidates current requirements into a technical specification format consistent with the Standard Technical Specifications and does not involve technical or plant modifications.

2) The proposed changes as described in Items 2.b.1, 2.b.2, 2.b.3 and 2.b.4 are similar to example (ii) of 48 FR 14870 in that they provide additional restrictions by: requiring OPERABILITY of all CONTAINMENT ISOLATION VALVES; adding SURVEILLANCE REQUIREMENTS that require a demonstration of valve OPERABILITY prior to returning a valve to service after maintenance and demonstration of valve OPERABILITY on actuation of Phase "A", "B", or Containment Ventilation Isolation test signal; and adding a table that lists CONTAINMENT ISOLATION VALVES.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.6.4 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: CONTAINMENT SYSTEM - COMBUSTIBLE GAS CONTROL MONITORS

NO: 3/4.6.5

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Table 3.5-5 Item 12, Table 4.1-1 Item 36, Table 4.18-1 Item 11, and B4.18.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more complete than the current Technical Specification as follows:

1. The revision requires CHANNEL CALIBRATION of the monitors on a STAGGERED TEST BASIS which is more restrictive than the current requirements.

2. The revised Specification LCO is applicable in MODES 1 and 2 which is consistent with Standard Technical Specification. The current Technical Specification applicability does not explicitly specify LCO Mode.

3. The revision requires CHANNEL CHECK at least once per 12 hours. The current Technical Specification specifies CHANNEL CHECK once per shift. The revision is more specific in defining the time limit in which surveillance should be performed.

c. The revision relaxes the following current requirement:

The revised specification surveillances are applicable in MODES 1 and 2. The current Technical Specification implies surveillance applicability in MODES 1 throug 4.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed changes as described in Item 2.a are similar to example (i) of 48 FR 14870 in that they are administrative changes which consolidate current requirements into a technical specification format consistent with the Standard Technical Specifications and do not involve technical or plant modifications.

2) The proposed changes as described in Items 2.b.1, 2.b.2 and 2.b.3 are similar to example (ii) of 48 FR 14870 in that they provide additional restrictions by: requiring CHANNEL CALIBRATION of the monitors on a STAGGERED TEST BASIS; specifying applicable LCO Modes; and stating more specific surveillance intervals for CHANNEL CHECKS.

3) The proposed change to relax the number of surveillance Modes, does not involve a significant hazards consideration because this change would not:

   a. Involve a significant increase in the probability of or consequences of an accident previously evaluated. The Hydrogen Monitors are required for detection of Hydrogen buildup within the containment following a LOCA to allow operator action to reduce the hydrogen concentration below its flammable limit. The current footnote in current Table 4.1-1 Item 36 would imply

that surveillance requirements would be required in revised Technical Specification MODES 1 through 4. The probability of a LOCA requiring hydrogen monitors while in MODES 3 or 4 is significantly less than MODES 1 and 2 because of the short time the plant spends in MODES 3 and 4. In addition, the severity of a LOCA which could lead to significant hydrogen buildup is less likely in MODES 3 and 4 because plant parameters are not close to design limits as is the case in MODES 1 and 2. Based on these considerations, there would not be a significant increase in the probability of or the consequences of an accident previously evaluated. Finally, the proposed revision to only include applicability in MODES 1 and 2 is consistent with industry practice and is consistent with the Standard Technical Specifications.

b. Create the possibility of a new or different kind of accident from any previously analyzed because the proposed change introduces no new mode of plant operation nor involves a physical modification the plant.

c. Involve a significant reduction in a margin of safety because of the short operating time the plant spends in MODES 3 and 4 and the operating parameters are further from design limits.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.6.5 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: POST ACCIDENT CONTAINMENT VENT SYSTEM

NO: 3/4.6.6

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specifications 3.4.6 and 4.7.2.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more complete than the current Technical Specification as follows:

The revision requires carbon analysis of the System's filter within 31 days of obtaining a sample. The current Technical Specification allows 45 days.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which consolidates current requirements into a technical specification format consistent with the Standard Technical Specifications and does not involve technical or plant modifications.

2) The proposed change as described in Item 2.b is similar to example (ii) of 48 FR 14870 in that it provides an additional restriction on the time required for the carbon analysis of the System's filter.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.6.6 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: TURBINE CYCLE - SAFETY VALVES

NO: 3/4.7.1.1

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

   As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specifications 3.8.1a, Table 4.1-2 Item 7 and B3.8.

2) Proposed Condition of License:

   a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

   b. The revision is more complete than the current Technical Specification as follows:

      1. The proposed revision specifies the Main Steam Line Safety Valves lift settings and orifice sizes.

      2. The proposed revision ACTION statement requires that with one or more Main Steam Line Code Safety Valves inoperable, the unit be placed in HOT STANDBY within 6 hours if the Power Range Neutron Flux High Trip Setpoint is not reduced within 4 hours. The current Technical Specification allows unit operation up to 48 hours with an inoperable Main Steam Line Code Safety Valve.

   c. The revision relaxes the following current requirement:

      The proposed revision allows for operation with one or more inoperable Main Steam Line Code Safety Valve(s) at a reduced power level beyond the current Technical Specification 48 hour limit.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility

in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed changes as described in Item 2.a are similar to example (i) of 48 FR 14870 in that they are administrative changes which consolidate current requirements into a technical specification format consistent with the Standard Technical Specifications and do not involve technical or plant modifications.

2) The proposed changes as described in Items 2.b.1 and 2.b.2 are similar to example (ii) of 48 FR 14870 in that they provide additional information by including Main Steam Line Code Safety Valves lift settings and orifice size, and additional limitations on plant operation with an inoperable Main Steam Line Code Safety Valve.

3) The proposed change to relax unit operation with one or more inoperable Main Steam Line Code Safety valves, does not involve a significant hazards consideration because this change would not:

   a. Involve a significant increase in the probability of or consequences of an accident previously evaluated. The operability of the main steam line Code Safety Valves ensures overpressure protection of system components during the most severe anticipated transient of turbine trip from 100% rated thermal power coincident with an assumed loss of condenser heat sink. Operation in MODES 1, 2 and 3 with inoperable safety valves is justified based on a reduction in secondary steam flow and thermal power consistent with the reduced reactor trip settings of the Power Range Neutron Flux channels. These reduced setpoints would ensure overpressure protection of system components with inoperable safety valves.

The revised technical specification allowing operation at reduced power levels with inoperable secondary safety valves is consistent with industry practice and Standard Technical Specifications.

b. Create the possibility of a new or different kind of accident from any previously analyzed because the proposed change introduces no new mode of plant operation nor involves a physical modification to the plant. Operation at a reduced power level is not a new mode of operation.

c. Involve a significant reduction in a margin of safety because the proposed revision requires reactor thermal power level to be reduced to accommodate the reduced Power Range Neutron Flux Setpoints. This reduction of thermal power level will ensure that the required margin for steam relief capacity is always within the number of OPERABLE SAFETY VALVES.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.7.1.1 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: PLANT SYSTEMS - AUXILIARY FEEDWATER SYSTEM

NO: 3/4.7.1.2

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 4.10 and Table 4.18-1, Item 3, and the Proposed Licensing Amendment transmitted to the NRC by Letter L-86-193 dated May 7, 1986 in proposed Specifications 3.18 and B3.18.

2) Proposed Condition of License:

a. The amendment consolidates the current surveillance requirements and the above referenced Proposed Licensing Amendment into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS consistent with the Standard Technical Specifications.

The footnote in the current Technical Specification that requires 600 gpm of AUXILIARY FEEDWATER flow to Unit 4 steam generators, prior to steam generator replacement, is no longer required because Unit 4 steam generators have been replaced. This change is considered administrative.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature.

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1) The proposed changes as described in Item 2.a are similar to example (i) of 48 FR 14870 in that they are administrative changes which consolidate current Surveillance requirements and the information in License Ammendment request L-86-193 into a technical specification format consistent with the Standard Technical Specifications and do not involve technical or plant modifications.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.7.1.2 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: PLANT SYSTEMS - CONDENSATE STORAGE TANKS

NO: 3/4.7.1.3

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the proposed Licensing Amendment Request (Letter L-86-193 dated May 7, 1986) to the Turkey Point Unit 3 and 4 Technical Specifications in proposed Specifications 3.19, B3.19 and 4.21.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature.

1) The proposed changes as described in Item 2.a are similar to example (i) of 48 FR 14870 in that they are administrative changes which consolidate requirements in License Amendment Request L-86-193 dated May 7, 1986 into a technical specification format consistent with the Standard Technical Specifications and do not involve technical or plant modifications.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.7.1.3 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is a reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: PLANT SYSTEMS - SPECIFIC ACTIVITY

NO: 3/4.7.1.4

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.8.2.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more complete than the current Technical Specification as follows:

1. The LCO specifies that the SPECIFIC ACTIVITY of secondary coolant be maintained less than or equal to 0.10 microcuries/gram DOSE EQUIVALENT I-131. The current Technical Specification requires that I-131 activity shall not exceed 0.67 microcuries/gram.

2. The action statement specifies plant shutdown to HOT STANDBY within 6 hours if the activity limit is exceeded. The current Technical Specification allows 48 hours for activity level reduction before shutdown action is taken.

3. The surveillance requirements for determination of Gross Radioactivity and isotopic analysis for DOSE EQUIVALENT I-131 concentration are specified. The current Technical Specification does not specify these requirements.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed changes as described in Item 2.a are similar to example (i) of 48 FR 14870 in that they are administrative changes which consolidate current requirements into a technical specification format consistent with the Standard Technical Specifications and do not involve technical or plant modifications.

2) The proposed changes as described in Items 2.b.1, 2.b.2 and 2.b.3 are similar to example (ii) of 48 FR 14870 in that they provide additional limitation, restrictions and control by specifying the activity of the secondary coolant be maintained less than or equal to 0.10 microcuries/gram DOSE EQUIVALENT I-131, more restrictive plant shutdown requirements if activity limit is exceeded and surveillance requirements for determination of Gross Radioactivity and isotopic analysis for DOSE EQUIVALENT I-131 concentration.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.7.1.4 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: PLANT SYSTEMS - MAIN STEAM LINE ISOLATION VALVES

NO: 3/4.7.1.5

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

   As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specifications 3.8.1.c and d, 4.9 and B4.9.

2) Proposed Condition of License:

   a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

   b. The revision is more complete than the current Technical Specification as follows:

      1) The proposed ACTION statements specify that in MODE 1 with one MSIV inoperable but open, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 4 hours; otherwise, power shall be reduced to less than or equal to 5% of RATED THERMAL POWER within the next 6 hours. In MODES 2 and 3 with one MSIV inoperable subsequent operation may proceed provided the isolation valve is maintained closed; otherwise, the unit shall be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. The current Technical Specification specifies 48 hours for restoration of MSIV to operable status before shutdown action is taken.

      2) The surveillance requirement specifies verification of MSIV operability per specification 4.0.5. The IST program requires closure time testing of these valves every COLD SHUTDOWN. The current Technical Specification specifies closure time testing once per refueling.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed changes as described in Item 2.a are similar to example (i) of 48 FR 14870 in that they are administrative changes which consolidate current requirements into a technical specification format consistent with the Standard Technical Specifications and do not involve technical or plant modifications.

2) The proposed changes as described in Items 2.b.1 and 2.b.2 are similar to example (ii) of 48 FR 14870 in that they provide additional restrictions. The revised ACTION statement provides a more restrictive allowed operation time with an INOPERABLE MSIV. The revised SURVEILLANCE REQUIREMENT provides for a more restrictive MSIV closure time test frequency.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.7.1.5 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: PLANT SYSTEMS - STANDBY FEEDWATER SYSTEM

NO: 3/4.7.1.6

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the Current Turkey Point Unit 3 and 4 Technical Specifications in Specifications 3.20, 4.21, B3.20 and B4.21.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION LIMITS and SURVEILLANCE REQUIREMENTS.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature.

1) The proposed changes as described in Item 2.a are similar to example (i) of 48 FR 14870 in that they are administrative changes which consolidated the current requirements into a technical specification format consistent with the Standard Technical Specifications and do not involve technical or plant modifications.

Proposed Tech. Spec. No. 3/4.7.1.6

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.7.1.6 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

NO: 3/4.7.2

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

The current Turkey Point Unit 3 and 4 Technical Specification does not specify requirements for STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATIONS.

2) Proposed Condition of License:

a. The revision is more complete than the current Technical Specification as follows:

The new requirements are specified in this proposed specification containing LCO, APPLICABILITY MODES, ACTION statements, and SURVEILLANCE REQUIREMENTS.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed changes as described in Item 2.a are similar to example (ii) of 48 FR 14870 in that they provide additional restrictions limitations and controls by including a new Technical Specification for steam generator pressure and temperature limitations.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.7.2 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: COMPONENT COOLING WATER SYSTEM

NO: 3/4.7.3

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.4.4, Table 4.18-1 Item 6 and B3.4.4.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more complete than the current Technical Specification as follows:

1. The revision requires applicability in MODES 1, 2, 3 and 4. The current Technical Specification specifies applicability only for reactor criticality.

2. The revision adds requirements for operability of CCW headers.

3. A surveillance requirement is added that requires verification of system equipment operation on safety injection test signal or bus undervoltage signal.

4. The revised surveillance requirement requires system walkdown for valve alignment once per 31 days in MODES 1 through 4. The current Technical Specification requires a system walkdown when T_{avg} is greater than 540 F.

c. The revision relaxes the following current requirement:

1. The current Technical Specification requires operability of three CCW pumps. The revision requires operability of two CCW pumps from independent power supplies.

2. The current Technical Specification allows two CCW pumps and one heat exchanger to be inoperable up to 24 hours. The revision allows two CCW pumps and one heat exchanger to be inoperable up to 72 hours.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed changes as described in Item 2.a are similar to example (i) of 48 FR 14870 in that they are administrative changes which consolidate current requirements into a technical specification format consistent with the Standard Technical Specifications and do not involve technical or plant modifications.

2) The proposed changes as described in Items 2.b.1, 2.b.2, 2.b.3 and 2.b.4 are similar to example (ii) of 48 FR 14870 in that they provide additional restrictions and controls by: increasing the applicable MODES of operation; adding requirements for operability of CCW headers; verifying safety-related equipment actuates to its correct position on an SI test signal; and including a SURVEILLANCE REQUIREMENT that requires a system walkdown once per 31 days when operating in MODES 1 through 4.

3) The proposed changes in item 2.c do not involve a significant hazards consideration because these changes would not:

   a. Involve a significant increase in the probability of or consequences of an accident previously evaluated.

The revision requires operability of two CCW pumps powered from two independent power supplies in lieu of operability of three pumps required by current Technical Specification. The Turkey Point plant has three 100% capacity CCW pumps, two of these pumps are powered from independent power sources and the remaining pump is a standby pump normally powered from the "B" power source. The safety analysis requires one CCW pump and two heat exchangers for normal plant operation and during shutdown. Operability of two CCW pumps provides redundancy and assurance that at least one pump is OPERABLE assuming a single failure thus meeting the design intent of Section 9.2.2 of the Standard Review Plan.

The revision allows operation with one OPERABLE CCW pump and two CCW heat exchangers for up to 72 hours. The current Technical Specification allows operation with one OPERABLE CCW pump and two CCW heat exchangers for up to 24 hours. This additional 48 hours is not considered to significantly increase the probability that consequences of previously evaluated accident could not be mitigated because the following unlikely sequence events would be required: failure of the OPERABLE CCW pump and two heat exchangers and unavailability of the third pump during the additonal 48 hour period. Also, the 24 hour limit for continued operation may not provide an adequate time to effect repairs and place inoperable equipment back into service prior to placing the plant in a transient condition for achieving shutdown.

Finally, the revised technical specification rquirements of two independent powered OPERABLE CCW loops and the 72 hour one loop operability limit is consistent with current industry practices and the Standard Technical Specifications.

b. Create the possibility of a new or different kind of accident from any previously analyzed because the proposed change introduces no new mode of plant operation nor involves a physical modification to the plant.

c. Involve a significant reduction in a margin of safety because the redundant cooling capacity required by this proposed specification, assuming a single failure of one of the two OPERABLE CCW Pumps, ensures continued operation of safety-related equipment during normal and accident conditions as discussed in Item a above. The extended time for one train OPERABLE to 72 hours does not involve a significant reduction in the margin of safety because of the unlikely sequence of events that would be required during the additional 48 hours resulting in the loss of all CCW.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.7.3 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: INTAKE COOLING WATER SYSTEM

NO: 3/4.7.4

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specifications 3.4.5, Table 4.18-1 Item 7 and B3.4.5.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more complete than the current Technical Specification as follows:

1. The revision requires applicability in MODES 1, 2, 3 and 4. The current Technical Specification specifies applicability only for reactor criticality.

2. The revised surveillance requirement requires a system walkdown once per 31 days in MODES 1 through 4. The current Technical Specification requires system walkdown for valve alignment when T_{avg} is greater than 540 F.

3. A surveillance requirement is added that requires verification of automatic valve actuation on a safety injection test signal, and of automatic pump start on a safety injection test signal or bus undervoltage test signal.

c. The revision relaxes the following current requirement:

1. The current Technical Specification requires OPERABILITY of three ICW pumps. The revision requires OPERABILITY of two ICW pumps powered from independent power supplies.

2. The current Technical Specification allows one of the pumps to be inoperable up to 24 hours before shutdown action is taken. The revision allows one of the required pumps to be inoperable up to 72 hours before shutdown action is taken.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed changes as described in Item 2.a are similar to example (i) of 48 FR 14870 in that they are administrative changes which consolidate current requirements into a technical specification format consistent with the Standard Technical Specifications and do not involve technical or plant modifications.

2) The proposed changes as described in Items 2.b.1, 2.b.2 and 2.b.3 are similar to example (ii) of 48 FR 14870 in that they provide additional restrictions by: increasing the applicable MODES of operation; verifying correct automatic valve actuation and automatic pump start on a safety injection test signal or bus undervoltage test signal; and including a SURVEILLANCE REQUIREMENT that requires a system walkdown once per 31 days when operating in MODES 1 through 4.

3) The proposed changes in Item 2c do not involve a significant hazards consideration because these changes would not:

   a. Involve a significant increase in the probability of or consequences of an accident previously evaluated.

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The revision requires OPERABILITY of two ICW pumps powered from two independent power supplies in lieu of OPERABILITY of three pumps required by current Technical Specification. The basis for this change is that the two ICW pumps will ensure that sufficient cooling capacity is available for continued operation of safety related equipment during normal and accident conditions assuming a single failure. One ICW pump is capable of providing the required cooling water flow to safety related loads.

The revision allows one of the two required ICW pumps and headers to be inoperable up to 72 hours before shutdown action is taken. This proposed inoperable allowed time is consistent with current industry practices and Standard Technical Specification. The proposed inoperable allowed time for ICW pump and header is consistent with inoperable allowed time of CCW pump and heat exchanger. The loss of redundant equipment is extremely remote as it would require loss of offsite power, the inability of the diesel generator to power the OPERABLE train and the third ICW pump as a backup ICW pump also being inoperable during the 72 hours.

b. Create the possibility of a new or different kind of accident from any previously analyzed because the proposed change introduces no new mode of plant operation nor involves a physical modification to the plant.

c. Involve a significant reduction in a margin of safety because the redundant cooling capacity required by this proposed specification, assuming a single failure, ensures continued operation of safety-related equipment during normal and accident conditions.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.7.4 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: ULTIMATE HEAT SINK

NO: 3/4.7.5

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

The current Turkey Point Unit 3 and 4 Technical Specification does not specify requirements for ULTIMATE HEAT SINK.

2) Proposed Condition of License:

a. The revision is more complete than the current Technical Specification as follows:

The amendment adds requirements for ULTIMATE HEAT SINK including LCO, APPLICABILITY MODES, ACTION statement, and SURVEILLANCE REQUIREMENTS.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed change as described in Item 2.a is similar to example (ii) of 48 FR 14870 in that it provides additional restrictions by including a new technical specification for the ULTIMATE HEAT SINK.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.7.5 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: FLOOD PROTECTION

NO: 3/4.7.6

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

The current Turkey Point Unit 3 and 4 Technical Specification does not specify requirements for FLOOD PROTECTION.

2) Proposed Condition of License:

a. The revision is more complete than the current Technical Specification as follows:

The amendments add requirements for FLOOD PROTECTION including LCO, APPLICABILITY MODES, ACTION statement, and SURVEILLANCE REQUIREMENTS.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed change as described in Item 2.a is similar to example (ii) of 48 FR 14870 in that it provides additional restrictions by including a new technical specification for FLOOD PROTECTION.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.7.5 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: CONTROL ROOM EMERGENCY AIR CLEANUP

NO: 3/4.7.7

A. DESCRIPTION OF CHANGES

1) Present condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specifications 3.4.7, 4.7.3 and B4.7.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more complete than the current Technical Specification as follows:

1. The revision specifies LCO applicability in all modes. The current Technical Specification specifies applicability before the reactor is made critical and provides exception during low power physics tests.

2. The revision allows 72 hours of system inoperability prior to shutdown. The current Technical Specification allows 3-1/2 days.

3. The revision ACTION statement for MODES 5 and 6 include suspension of core alterations or positive reactivity changes. The current Technical Specification has no similar requirements.

4. The revision adds a SURVEILLANCE REQUIREMENT that requires verification of control room temperature.

5. The revision requires carbon analysis of the system's filter within 31 days of obtaining a sample. Current Technical Specification allows 45 days.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed changes as described in Item 2a are similar to example (i) of 48 FR 14870 in that they are administrative changes which consolidate current requirements into a technical specification format consistent with the Standard Technical Specification.

2) The proposed changes as described in Item 2.b are similar to example (ii) of 48 FR 14870 in that they provide additional restrictions and controls by: specifying LCO applicability for all operational MODES; allowing less system inoperability time; adding an ACTION statement for MODES 5 and 6 and a SURVEILLANCE REQUIREMENT that verifies the control room temperature and requiring a more frequent carbon analysis of the system's filters.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.7.7 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: SNUBBERS

NO: 3/4.7.8

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the Turkey Point Unit 3 and 4 Technical Specification in Specifications 3.13 and 4.14.

2) Proposed Condition of License:

a. The Amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature.

1) The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which consolidates current requirements into a technical specification format consistent with the Standard Technical Specifications and does not involve technical or plant modifications.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.7.8 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: SEALED SOURCE CONTAMINATION

NO: 3/4.7.9

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.11, 4.13 and B3.11.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more complete than the current Technical Specification as follows:

1. The LCO requires that SEALED SOURCE containing radioactive material in excess of 100 microcuries of beta and/or gamma emitting material or 5 microcuries of alpha emitting material shall be free of $\geq$ 0.005 microcurie of removable contamination. The current Technical Specification references 10 CFR 30.71 Schedule B for by-product material for leak testing. 10 CFR 30.71 contains by-products that are not applicable to nuclear power plant.

2. The revision requires a submittal of a report to the Commission if SEALED SOURCE or Fission Detector Leakage tests reveal the presence of removable contamination greater than allowed by LCO. The current Technical Specification does not address this reporting requirement.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which consolidates current requirements into a technical specification format consistent with the Standard Technical Specifications and does not involve technical or plant modifications.

2) The proposed changes as described in Items 2.b.1 and 2.b.2 are similar to example (ii) of 48 FR 14870 in that they provide additional restrictions by specifying SEALED SOURCE CONTAMINATION limits in the LCO and requiring a submittal of a report to the Commission if a SEAL SOURCE or Fission Detector Leakage test reveals the presence of a removable contamination greater than allowed by the LCO.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.7.9 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: REFUELING OPERATIONS - BORON CONCENTRATION

NO: 3/4.9.1

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

   As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.10.8, Table 4.1-2 Item 13 and B3.10.8.

2) Proposed Condition of License:

   a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

   b. The revision is more complete than the current Technical Specification as follows:

      1. The addition rate of Boric Acid Solution required by the ACTION statement has been increased to be consistent with the boron addition rates required in Specification 3/4.1.1.2.

      2. The revision adds SURVEILLANCE REQUIREMENT 4.9.1.1 that requires determination of reactivity conditions prior to removing or unbolting the reactor vessel head, and withdrawal of any full length control rod in excess of 3 feet.

      3. The revision adds SURVEILLANCE REQUIREMENT 4.9.1.3 that requires verfication that primary water supply to the boric acid blender is closed.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which consolidates current requirements into a technical specification format consistent with the Standard Technical Specifications and does not involve technical or plant modifications.

2) The proposed changes as described in Item 2.b.1, 2.b.2 and 2.b.3 are similar to example (ii) of 48 FR 14870 in that they provide additional limitations and controls by including the increased boron addition rate consistent with proposed specifications 3/4.1.1.2 and additional surveillance requirements relating to valve line ups and boron concentration monitoring prior to head removal.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.9.1 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: REFUELING OPERATIONS - INSTRUMENTATION

NO: 3/4.9.2

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.10.3, Table 4.1-1 Item 3 and B3.10.3.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more complete than the current Technical Specification as follows:

1. The LCO required operability of audible indication associated with Source Range Monitors.

2. The revision adds SURVEILLANCE REQUIREMENTS that requires Analog Channel Operational Test of source range monitors at least once per 7 days.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which consolidates current requirements into a technical specification format consistent with the Standard Technical Specifications and does not involve technical or plant modifications.

2) The proposed changes as described in Items 2.b.1 and 2.b.2 are similar to example (ii) of 48 FR 14870 in that they provide additional controls for source range monitors by including audible alarms and additional surveillance testing on a weekly basis for ANALOG CHANNEL OPERATIONAL TEST.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.9.2 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: REFUELING OPERATIONS - DECAY TIME

NO: 3/4.9.3

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.10.5 and B3.10.5.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more complete than the current Technical Specification as follows:

Adds additional SURVEILLANCE REQUIREMENTS to verify that the reactor has been subscritical for at least 100 hours prior to movement of irradiated fuel.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which consolidates current requirements into a technical specification format consistent with the Standard Technical Specifications and does not involve technical or plant modifications.

2) The proposed change as described in Item 2.b is similar to example (ii) of 48 FR 14870 in that it provides additional controls by including the surveillance which verifies the time since reactor shutdown.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.9.3 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: REFUELING OPERATIONS - CONTAINMENT BUILDING PENETRATIONS

NO: 3/4.9.4

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.10.1 AND B3.10.1.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more complete than the current Technical Specification as follows:

The revision adds SURVEILLANCE REQUIREMENTS for verification of closure of containment building penetrations.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which consolidates current requirements into a technical specification format consistent with the Standard Technical Specifications and does not involve technical or plant modifications.

2) The proposed change as described in Item 2.b is similar to example (ii) of 48 FR 14870 in that it provides additional controls by including the surveillance requirements for verification of closure of containment building penetrations.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.9.4 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: REFUELING OPERATIONS - COMMUNICATIONS

NO: 3/4.9.5

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

   As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.10.6 and B3.10.6.

2) Proposed Condition of License:

   a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

   b. The revision is more complete than the current Technical Specification as follows:

      The Revision adds SURVEILLANCE REQUIREMENT for periodic verification of operation of communication between the control room and personnel at the refueling station.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which consolidates current requirements into a technical specification format consistent with the Standard Technical Specifications and does not involve technical or plant modifications.

2) The proposed change as described in Item 2.b is similar to example (ii) of 48 FR 14870 in that it provides additional controls by including the surveillance which verifies communication between the control room and the refueling station.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.9.5 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: REFUELING OPERATIONS - MANIPULATOR CRANE

NO: 3/4.9.6

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification Table 4.1-2 Item 9.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more complete than the current Technical Specification as follows:

1. Manipulator crane load capacity and overload cutoff set point have been added to the specification.

2. Auxiliary crane load capacity has been added to the specification.

3. LCO applicability and action statements have been added to the specification.

4. The time interval between surveillance testing and start of refueling operation was specified as 100 hours.

5. A load test of both the manipulator crane and the auxiliary hoist has been added to the surveillance testing.

6. Specific SURVEILLANCE REQUIREMENTS for retesting of the manipulator crane and the auxiliary hoist have been added.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which consolidates current requirements into a technical specification format consistent with the Standard Technical Specifications and does not involve technical or plant modifications.

2) The proposed changes as described in Items 2.b.1, 2.b.2, 2.b.3, 2.b.4, 2.b.5 and 2.b.6 are similar to example (ii) of 48 FR 14870 in that they provide additional limitations and restrictions by including the manipulator crane and auxiliary hoist load ratings and manipulator crane overload cutoff point, LCO and ACTION statements, maximum time interval between test and fuel handling operation and additional surveillance load testing to the operability criteria.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.9.6 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: CRANE TRAVEL - SPENT FUEL STORAGE AREAS

NO: 3/4.9.7

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.10.9 and B3.10.9.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more complete than the current Technical Specification as follows:

1. The revision assures that only one fuel assembly will be handled at a time over the spent fuel pit.

2. The revision adds a surveillance requirement that requires verification of total load on the spent fuel crane.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

Proposed Tech. Spec. No. 3/4.9.7

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed changes as described in Item 2.a is similar to example (i) of 48 FR 14870 in that they are administrative changes which consolidate current requirements into a technical specification format consistent with the Standard Technical Specifications and do not involve technical or plant modifications.

2) The proposed changes as described in Items 2.b.1 and 2.b.2 are similar to example (ii) of 48 FR 14870 in that they provide additional limitations and controls by including the requirement to handle only one fuel element at a time over the spent fuel pit and a requirement that the total load on the spent fuel crane be determined and compared to the limit in the LCO.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.9.7 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION - HIGH WATER LEVEL

NO: 3/4.9.8.1

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.10.7.1, Table 4.1-2 Item 18 and B3.10.7.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more complete than the current Technical Specification as follows:

The surveillance requirement in the revision specifies verification that RHR loop circulation flow is at least 3000 gpm. The current Technical Specification specifies verification of flow but does not specify a value. The circulation flow which is an alarmed parameter has been exchanged for core outlet temperature which does not alarm in the control room.

c. The revision relaxes the following current requirement:

The frequency of monitoring the RHR cooling system operation has been decreased from every 4 hours to every 12 hours.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which consolidates current requirements into a technical specification format consistent with the Standard Technical Specifications and does not involve technical or plant modifications.

2) The proposed change as described in Item 2.b is similar to example (ii) of 48 FR 14870 in that it provides additional information by including the required minimum flow from the RHR cooling loop which is an alarmed parameter.

3) The proposed change to relax the time interval for checking RHR loop cooling operability does not involve a significant hazards consideration because this change would not:

   a. Involve a significant increase in the probability of or consequences of an accident previously evaluated. By exchanging the temperature measurement for the alarmed flow measurement to determine operability of the required RHR loop the probability of losing RHR cooling without being noticed by the operator is less. The low flow alarm will alert the operator to investigate and restore cooling. Increasing the surveillance time interval is justified by the continuous monitor provided by the low flow alarm.

   b. Create the possibility of a new or different kind of accident from any previously analyzed because the proposed change introduces no new mode of plant operation nor involves a physical modification to the plant.

   c. Involve a significant reduction in a margin of safety because one method of monitoring RHR cooling capability is being exchanged for another and is consistent with industry practice in that it is the same as Standard Technical Specifications.

Proposed Tech. Spec. No. 3/4.9.8.1

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.9.8.1 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION - LOW WATER LEVEL

NO: 3/4.9.8.2

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.10.7.2, Table 4.1-2 Item 18 and B3.10.7.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more complete than the current Technical Specification as follows:

The surveillance requirement in the revision specifies verification that RHR loop circulation flow is at least 3000 gpm. The current Technical Specification specifies verification of flow but does not specify a value. The circulation flow which is an alarmed parameter has been exchanged for core outlet temperature which does not alarm in the control room.

c. The revision relaxes the following current requirement:

The frequency of monitoring the RHR cooling system operation has been decreased from every 4 hours to every 12 hours.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which consolidates current requirements into a technical specification format consistent with the Standard Technical Specifications and does not involve technical or plant modifications.

2) The proposed change as described in Item 2.b is similar to example (ii) of 48 FR 14870 in that it provides additional information by including the required minimum flow from the RHR cooling loop which is an alarmed parameter.

3) The proposed change to relax the time interval for checking RHR loop cooling operability does not involve a significant hazards consideration because this change would not:

   a) Involve a significant increase in the probability of or consequence of an accident previously evaluated. By exchanging the temperature measurement for the alarmed flow measurement to determine operability of the required RHR loop the probability of losing RHR cooling without being noticed by the operator is less. The low flow alarm will alert the operator to investigate and restore cooling. Increasing the surveillance time interval is justified by the continous monitor provided by the low flow alarm.

   b. Create the possibility of a new or different kind of accident from any previously analyzed because the proposed change introduces no new mode of plant operation nor involves a physical modification to the plant.

   c. Involve a significant reduction in a margin of safety because one method of monitoring RHR cooling capability is being exchanged for another and is consistent with industry practice in that it is the same as Standard Technical Specifications.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.9.8.2 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: REFUELING OPERATIONS - CONTAINMENT VENTILATION ISOLATION SYSTEM

NO: 3/4.9.9

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.10.2, Table 4.1-2 Item 8, B3.10.2.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more complete than the current Technical Specification as follows:

A Surveillance requirement has been added to verify that the Containment Ventilation Isolation System is operable within 100 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which consolidates current requirements into a technical specification format consistent with the Standard Technical Specifications and does not involve technical or plant modifications.

2) The proposed change as described in Items 2.b is similar to example (ii) of 48 FR 14870 in that it provides additional controls by adding a surveillance requirement to check the operability of the Containment Ventilation Isolation System.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.9.9 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: REFUELING OPERATIONS - WATER LEVEL REACTOR VESSEL

NO: 3/4.9.10

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

The current Turkey Point Unit 3 and 4 Technical Specification does not explicitly specify requirements for WATER LEVEL - REACTOR VESSEL.

2) Proposed Condition of License:

a. The proposed revision provides a new technical specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The proposed revision is more complete than the current Technical Specification as follows:

A new Technical Specification is being proposed to be added consistent with Standard Technical Specifications. Addition of this specification will assure that proper water level is verified above the reactor vessel flange during refueling operation.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed change as described in Items 2.a and 2.b are similar to example (ii) of 48 FR 14870 in that they provide additional limitations, restrictions and controls by adding a technical specification for Reactor Vessel Water Level.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.9.10 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: REFUELING OPERATIONS - WATER LEVEL STORAGE POOL

NO: 3/4.9.11

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

The current Turkey Point Unit 3 and 4 Technical Specification does not explicitly specify requirements for WATER LEVEL - STORAGE POOL.

2) Proposed Condition of License:

a. The amendment consolidates the requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more complete than the current Technical Specification as follows:

This new Technical Specification is being proposed to be added. Addition of this specification will assure that proper water level is maintained and verified in the storage pool.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which consolidates current requirements into a technical specification format consistent with the Standard Technical Specifications and does not involve technical or plant modifications.

2) The proposed change as described in Item 2.b is similar to example (ii) of 48 FR 14870 in that it provides additional restrictions, limitations and controls by including a Technical Specification for STORAGE POOL WATER LEVEL.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.9.11 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: REFUELING - CASK HANDLING

NO: 3/4.9.12

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.12, Table 4.1-2 Item 17 and B3.12.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature.

1) The proposed changes as described in Item 2.a are similar to example (i) of 48 FR 14870 in that they are administrative changes which consolidate current requirements into a technical specification format consistent with the Standard Technical Specifications and do not involve technical or plant modifications.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.9.12 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: REFUELING OPERATIONS - RADIATION MONITORING

NO: 3/4.9.13

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.10.4, TABLE 4.1-1 Item 18A and B3.10.4.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more complete than the current Technical Specification as follows: -

1. Surveillance time interval for the CHANNEL CHECK has been shortened from daily to once per shift (see Table 4.3-3).

2. The CHANNEL CALIBRATION Surveillance has been added once per refueling.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which consolidates current requirements into a technical specification format consistent with the Standard Technical Specifications and does not involve technical or plant modifications.

2) The proposed changes as described in Items 2.b.1, and 2.b.2 are similar to example (ii) of 48 FR 14870 in that they provide additional restriction by adding a new surveillance for CHANNEL CALIBRATION once per refueling and reducing the CHANNEL CHECK surveillance time interval from daily to once per shift.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.9.13 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: REFUELING - SPENT FUEL STORAGE

NO: 3/4.9.14

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.17, Table 4.1-2 Item 13, B3.17.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more complete than the current Technical Specification as follows:

The revision adds an ACTION statement which is appropriate for the LCO requirements. The current Technical Specification does not specify ACTION statements.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which consolidates current requirements into a technical specification format consistent with the Standard Technical Specifications and does not involve technical or plant modifications.

2) The proposed change as described in Item 2.b is similar to example (ii) of 48 FR 14870 in that it provides additional restriction by including the ACTION STATEMENT in the revised technical specification.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.9.14 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: SPECIAL TEST EXCEPTIONS - SHUTDOWN MARGIN

NO: 3/4.10.1

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.2.1f.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more restrictive than the current Technical Specification as follows:

The SPECIAL TEST EXCEPTION to the SHUTDOWN MARGIN requirement is applicable to control rod worth and SHUTDOWN MARGIN measurements only. Also, SHUTDOWN reactivity equivalent to the highest worth control rod is required.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which consolidates current requirements into a technical specification format consistent with the Standard Technical Specifications and does not involve technical or plant modifications.

2) The proposed change as described in Item 2.b is similar to example (ii) of 48 FR 14870 in that it provides more restrictive exceptions to the SHUTDOWN MARGIN LCO by excluding only control rod worth and SHUTDOWN MARGIN measurements from the SHUTDOWN MARGIN requirements of Technical Specification 3.1.1.1.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.10.1 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: SPECIAL TEST EXCEPTIONS - GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

NO: 3/4.10.2

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.2.1a, b and c and 3.2.6.d and h.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more restrictive than the current Technical Specification as follows:

The special test exception to the control rod group height and insertion limits and selected power distribution limits requires that Power be limited to 85% or less.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed changes as described in Item 2.a are similar to example (i) of 48 FR 14870 in that they are administrative changes which consolidate current requirements into a technical specification format consistent with the Standard Technical Specifications and do not involve technical or plant modifications.

2) The proposed change as described in Item 2.b is similar to example (ii) of 48 FR 14870 in that it provides additional restrictions on core power (85% or less) during PHYSICS TESTS.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.10.2 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: SPECIAL TEST EXCEPTIONS - PHYSICS TESTS

NO: 3/4.10.3

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specifications 3.1.2.1 and 3.2.1.a, b and c.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more restrictive than the current Technical Specification as follows:

The RCS temperature is required to be 531 F or greater.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed changes as described in Item 2.a are similar to example (i) of 48 FR 14870 in that they are administrative changes which consolidate current requirements into a technical specification format consistent with the Standard Technical Specifications and do not involve technical or plant modifications.

2) The proposed change as described in Item 2.b is similar to example (ii) of 48 FR 14870 in that it provides an added restriction which requires a minimum RCS temperature of 531 F.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.10.3 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: SPECIAL TEST EXCEPTIONS - REACTOR COOLANT LOOPS

NO: 3/4.10.4

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

   As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.1.1.a.

2) Proposed Condition of License:

   a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature.

1) The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which consolidates current requirements into a technical specification format consistent with the Standard Technical Specifications and do not involve technical or plant modifications.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.10.4 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: SPECIAL TEST EXCEPTIONS - POSITION INDICATION SYSTEM - SHUTDOWN

NO: 3/4.10.5

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

There is no corresponding LCO requirement in the current Turkey Point Technical Specifications.

2) Proposed Condition of License:

a. The amendment adds a new Technical Specification that specifies the LCO, APPLICABLE MODES, ACTION statements and SURVEILLANCE REQUIREMENTS for SPECIAL TEST EXCEPTIONS during control rod drop time tests.

b. The revision is more restrictive than the current requirements as follows:

The special test exceptions to the Rod Position Indication System, for rod drop testing, restrict bank withdrawal to one bank and require the rod position indicator to be OPERABLE during rod withdrawal.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed changes as described in Item 2.a are similar to example (i) of 48 FR 14870 in that they are administrative changes which consolidate current requirements into a technical specification format consistent with the Standard Technical Specifications and do not involve technical or plant modifications.

2) The proposed change as described in Item 2.b is similar to example (ii) of 48 FR 14870 in that it provides added restrictions and controls that restrict rod bank withdrawal to one bank and requires the rod position indicator to be OPERABLE during rod withdrawal for the rod drop tests.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.10.5 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: LIQUID EFFLUENTS - CONCENTRATION

NO: 3/4.11.1.1

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.9, 3.9.1a, Table 3.9-1 and B3.9.1a.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more complete than the current Technical Specification as follows:

The proposed revision does not allow the use of gross beta-gamma analysis in lieu of isotopic analysis as a basis for liquid releases. (See Table 4.11-1).

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which consolidates current requirements into a technical specification format consistent with the Standard Technical Specifications and does not involve technical or plant modifications.

2) The proposed change as described in Item 2.b is similar to example (ii) of 48 FR 14870 in that it provides additional restrictions by allowing only isotopic analysis as a basis for liquid releases.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.11.1.1 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: LIQUID EFFLUENTS - DOSE

NO: 3/4.11.1.2

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

   As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.9, 3.9.1.b, 6.9.3e and B3.9.1.b.

2) Proposed Condition of License:

   a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature.

1) The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which consolidates current requirements into a technical specification format consistent with the Standard Technical Specifications and does not involve technical or plant modifications.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.11.1.2 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: LIQUID RADWASTE TREATMENT SYSTEM

NO: 3/4.11.1.3

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.9, 3.9.1.d, B3.9.1.d and 6.9.3.f.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

The revision clearly states that the release of radioactive materials be calculated for each unit. The bases section provides a method of calculating releases from each unit for a shared system.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature.

1) The proposed changes are similar to example (i) of 48 FR 14870 Items 2a are administrative changes which consolidate current requirements into a technical specification format consistent with the Standard Technical Specifications and do not involve technical or plant modifications. The proposed change as described is an administrative change in that it corrects a typographical error by including the requirement to calculate the release of radioactive materials for each unit when performing the surveillance for this specification.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.11.1.3 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: GASEOUS EFFLUENTS - DOSE RATE

NO: 3/4.11.2.1

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

   As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.9, 3.9.2a, Table 3.9-3 and B3.9.2.a.

2) Proposed Condition of License:

   a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

      Item D of Table 3.9-3 (current Technical Specification) has been reformated into two explicit requirements which more clearly represent the plants as-built configurations.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature.

1) The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which consolidates current requirements into a technical specification format consistent with the Standard Technical Specifications and does not involve technical or plant modifications.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.11.2.1 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: RADIOACTIVE EFFLUENTS/DOSE - NOBLE GASES

NO: 3/4.11.2.2

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.9, 3.9.2b, B3.9.2b and 6.9.3.e.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature.

1) The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which consolidates current requirements into a technical specification format consistent with the Standard Technical Specifications and does not involve technical or plant modifications.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.11.2.2 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: DOSE - I-131, I-133, TRITIUM, AND RADIOACTIVE MATERIAL IN PARTICULATE FORM

NO: 3/4.11.2.3

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.9, 3.9.2.c, B3.9.2.c and 6.9.3.e.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature.

1) The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which consolidates current requirements into a technical specification format consistent with the Standard Technical Specifications and does not involve technical or plant modifications.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.11.2.3 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: GASEOUS RADWASTE TREATMENT SYSTEM

NO: 3/4.11.2.4

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.9, 3.9.2e, B3.9.2e and 6.9.3g.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature.

1) The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which consolidates current requirements into a technical specification format consistent with the Standard Technical Specifications and does not involve technical or plant modifications.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.11.2.4 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: EXPLOSIVE GAS MIXTURES

NO: 3/4.11.2.5

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.9, 3.9.2.g, Table 3.9-4 and B3.9.2.g.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more complete than the current Technical Specification as follows:

The statement permitting release of the tank contents as a method of bringing the GAS DECAY TANKS into specification has been deleted.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which consolidates current requirements into a technical specification format consistent with the Standard Technical Specifications and does not involve technical or plant modifications.

2) The proposed change as described in Item 2.b is similar to example (ii) of 48 FR 14870 in that it provides additional limitations by deleting the specific option of releasing the tank if specifications are exceeded under the premise that any safe means can be used to bring the GAS DECAY TANKS into specification compliance.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.11.2.5 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: GAS DECAY TANK

NO: 3/4.11.2.6

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.9, 3.9.2.f and B3.9.2.f.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more complete than the current Technical Specification as follows:

The revision requires that whenever the radioactive material limit in the tank is exceeded, the event should be described in the next semiannual radioactive effluent release report.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which consolidates current requirements into a technical specification format consistent with the Standard Technical Specifications and does not involve technical or plant modifications.

2) The proposed change as described in Item 2.b is similar to example (ii) of 48 FR 14870 in that it provides additional restriction by requiring the event to be described in the semiannual report if the specification limit is exceeded.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.11.2.6 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: SOLID RADIOACTIVE WASTES

NO: 3/4.11.3

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.9.3 and B3.9.3.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

The current Technical Specification requirements to test bead resins for free standing liquids are not explicitly stated in the Revised Technical Specifications inorder to conform to the wording in the Standard Technical Specifications. This is considered an administrative change in that the requirement in the Revised Technical Specification to dewater radioactive waste in accordance with the PROCESS CONTROL PROGRAM provides the same requirement to assure that bead resin is dewatered properly.

b. The revision is more complete than the current Technical Specification in that SOLIDIFICATION and dewatering are addressed consistent with the Standard Technical Specification.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which consolidates current requirements into a technical specification format consistent with the Standard Technical Specifications and does not involve technical or plant modifications.

2) The proposed change as described in Item 2b is similar to example (ii) of 48 FR 14870 in that it provides additional controls by stating LCO ACTION and SURVEILLANCE REQUIREMENTS for SOLIDIFICATION and dewatering in accordance with the Standard Technical Specification.

Based on the above considerations the changes included in the development. of proposed Technical Specification 3/4.11.3 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: RADIOACTIVE EFFLUENTS - TOTAL DOSE

NO: 3/4.11.4

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 3.9, 3.9.2.h, B3.9.2.h and 6.9.3.h.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into this specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

b. The revision is more complete than the current Technical Specification as follows:

The revision adds a surveillance requirement 4.11.4.2 that requires determination of dose contributions from direct radiation if release rates exceed twice the limits of other specific radioactive effluent specifications.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which consolidates current requirements into a technical specification format consistent with the Standard Technical Specifications and does not involve technical or plant modifications.

2) The proposed change as described in Item 2.b is similar to example (ii) of 48 FR 14870 in that it provides additional controls adding a surveillance requiring that direct radiation from the plant be included in offsite dose calculations when certain release limits are exceeded.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.11.4 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: RADIOLOGICAL ENVIRONMENTAL MONITORING - MONITORING PROGRAM

NO: 3/4.12.1

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 4.12.1, B4.12.1 and 6.9.3i.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into the proposed specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature.

1) The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which consolidates current requirements into a technical specification format consistent with the Standard Technical Specifications and does not involve technical or plant modifications.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.12.1 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: RADIOLOGICAL ENVIRONMENTAL MONITORING - LAND USE CENSUS

NO: 3/4.12.2

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specifications 4.12.2 and B4.12.2.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into the proposed specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature.

1) The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which consolidates current requirements into a technical specification format consistent with the Standard Technical Specifications and does not involve technical or plant modifications.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.12.2 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: RADIOLOGICAL ENVIRONMENTAL MONITORING - INTERLABORATORY COMPARISON PROGRAM

NO: 3/4.12.3

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 4.12.3 and B4.12.3.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into the proposed specification and explicitly states the LCO, APPLICABLE MODES, ACTION Limits and SURVEILLANCE REQUIREMENTS.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature.

1) The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which consolidates current requirements into a technical specification format consistent with the Standard Technical Specifications and does not involve technical or plant modifications.

Based on the above considerations the changes included in the development of proposed Technical Specification 3/4.12.3 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: DESIGN FEATURES - SITE

NO: 5.1

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 5.1.

2) Proposed Condition of License:

a. The amendment reformats the SITE features depicted in the current Technical Specification for consistency with Standard Technical Specification.

b. The revision is more complete than the current technical specification as follows:

The LOW POPULATION ZONE area is identified on Figure 5.1-1.

c. The revision relaxes the following current requirement.

The exclusion area boundary is changed to be consistent with the safety analysis.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed changes as described in Item 2.a are similar to example (i) of 48 FR 14870 in that they are administrative changes which consolidate current requirements into a technical specification format consistent with the Standard Technical Specifications and do not involve technical or plant modifications.

2) The proposed changes as described in Item 2.b are similar to example (ii) of 48 FR 14870 in that they provide additional information by including the LOW POPULATION ZONE in the revised technical specification figure.

3) The proposed change to relax the exclusion area boundaries does not involve a significant hazards consideration because this change would not:

   a. Involve a significant increase in the probability of or consequences of an accident previously evaluated. The acceptance criteria for Standard Review Plan 15.6.5 recommends that the distances to the exclusion area boundary and to the low population zone outer boundary are acceptable if the total calculated radiological consequences (i.e., thyroid and whole body doses) for the hypothetical LOCA fall within the appropriate exposure guideline values specified in 10 CFR Part 100. The exclusion area boundary as depicted in the proposed Figure 5.1-1 is based on safety analyses that shows that the dose releases to the environment at the exclusion area boundary in the event of a LOCA are substantially less than the guidelines specified in 10 CFR Part 100.

   b. Create the possibility of a new or different kind of accident from any previously analyzed because the proposed change introduces no new mode of plant operation nor involves a physical modification to the plant.

   c. Involve a significant reduction in a margin of safety because the dose releases to the environment at the revised exclusion area boundary is as described in Chapter 14 of the FSAR, is based on existing safety analysis, and are substantially less than the guidelines specified in 10 CFR Part 100.

Based on the above considerations the changes included in the development of proposed technical specification 5.1 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: CONTAINMENT - CONFIGURATION

NO: 5.2.1

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

The current Turkey Point Unit 3 and 4 Technical Specification does not specify Containment - Configuration design features.

2) Proposed Condition of License:

a. The revision is more complete then the current Technical Specification as follows:

Containment design features are specified for consistency with the Standard Technical Specification.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards considerations are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example a more stringent surveillance requirement.

1) The proposed changes as described in Item 2a are similar to example (ii) of 48 FR 14870 in that they provide additional information by including Containment design features.

Based on the above considerations the changes included in the development of proposed Technical Specification 5.2.1 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: CONTAINMENT DESIGN PRESSURE AND TEMPERATURE

NO: 5.2.2

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification 5.3.A.2.

2) Proposed Condition of License:

a. The amendment reformats the design features identified in the current Technical Specification for consistency with Standard Technical Specification. The containment structural loads are covered by proposed Specification 5.2.3.

b. The revision is more complete than the current requirements as follows:

The maximum design internal containment temperature is specified.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards considerations are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the technical specifications for example, a more stringent surveillance requirement.

1) The proposed changes as described in Item 2.a are similar to example (i) of 48 FR 14870 in that they are administrative changes which consolidate current requirements into a technical specification format consistent with the Standard Technical Specifications and do not involve technical or plant modifications.

2) The proposed changes as described in Item 2.b are similar to example (ii) of 48 FR 14870 in that they provide additional information by including the maximum internal containment temperature in the proposed Technical Specification.

Based on the above considerations the changes included in the development of proposed Technical Specification 5.2.2 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: CONTAINMENT STRUCTURAL LOADS

NO: 5.2.3

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 5.3.A.2.

2) Proposed Condition of License:

a. The amendment reformats the structural loads design features described in the current Technical Specification into the proposed specification.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards considerations are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature.

1) The changes in Item 2.a are similar to example (i) of 48 FR 14870 in that they are administrative changes which reformat the structural loads design features described in the current Technical Specification into the proposed specification.

Based on the above considerations the changes included in the development of proposed Technical Specification 5.2.3 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: CONTAINMENT - PENETRATIONS

NO: 5.2.4

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 5.3.B.

2) Proposed Condition of License:

a. The amendment reformats the CONTAINMENT-PENETRATIONS design features described in the current Technical Specification into the proposed specification.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards considerations are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature.

1) The proposed changes described in Item 2.a are similar to example (i) of 48 FR 14870 in that they are administrative changes which reformat the CONTAINMENT-PENETRATIONS design features described in the current Technical Specification into the proposed specification.

Based on the above considerations the changes included in the development of proposed Technical Specification 5.2.4 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: CONTAINMENT SYSTEMS

NO: 5.2.5

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 5.3.C.

2) Proposed Condition of License:

a. The amendment reformats the design features stated in the current Technical Specification into the proposed specification.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards considerations are included in 10 CFR 50.92. The regulations state that if operation of facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature.

1) The proposed changes described in Item 2.a are similar to example (i) of 48 FR 14870 in that they are administrative changes which reformat the design features stated in the current Technical Specification into the proposed specification.

Based on the above considerations the changes included in the development of proposed Technical Specification 5.2.5 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: CONTAINMENT FUNCTION

NO: 5.2.6

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 5.3.A.1.

2) Proposed Condition of License:

a. The proposed revision reformats the design features stated in the current Technical Specification into the proposed specification.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards considerations are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature.

1) The proposed changes described in Item 2.a are similar to example (i) of 48 FR 14870 in that they are administrative changes which reformat the design features stated in the current Technical Specification into the proposed specification.

Based on the above considerations the changes included in the development of proposed Technical Specification 5.2.6 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: FUEL ASSEMBLIES

NO: 5.3.1

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 5.2.1.

2) Proposed Condition of License:

a. The amendment reformats the design features depicted in the current Technical Specification into the proposed specification.

b. The revision is more complete than the current Technical Specification as follows:

The nominal active fuel rod length is specified.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards considerations are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which consolidates current requirements into a technical specification format consistent with the Standard Technical Specifications and does not involve technical or plant modifications.

2) The proposed change as described in Item 2.b is similar to example (ii) of 48 FR 14870 in that it provides an additional design feature by including the nominal active fuel rod length.

Based on the above considerations the changes included in the development of proposed Technical Specification 5.3.1 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: CONTROL ROD ASSEMBLIES

NO: 5.3.2

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 5.2.5.

2) Proposed Condition of License:

a. The proposed amendment reformats the design features described in the current Technical Specification for consistency with the Standard Technical Specification. The reference to partial-length RCC has been deleted as these RCC assemblies have been removed from the reactor. The nominal length of the control rod assemblies has been corrected.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards considerations are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature.

1) The proposed changes as described in Item 2.a are similar to example (i) of 48 FR 14870 in that they are administrative changes which consolidate current requirements into a technical specification format consistent with the Westinghouse Standard Technical Specifications and do not involve technical or plant modifications. The deletion of the reference to partial length RCC assemblies is considered an administrative change because it is consistent with the footnote in the current technical specification.

Based on the above considerations the changes included in the development of proposed Technical Specification 5.3.2 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: BURNABLE POISON ROD ASSEMBLIES

NO: 5.3.3

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 5.2.4.

2) Proposed Condition of License:

a. The amendment reformats the design features stated in the current Technical Specification into the proposed specification.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards considerations are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature.

1) The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which reformats the BURNABLE POISON ROD ASSEMBLIES design features described in the current Technical Specification.

Based on the above considerations the changes included in the development of proposed Technical Specification 5.3.3 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: RCS - DESIGN PRESSURE AND TEMPERATURE

NO: 5.4.1

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

The current Turkey Point Unit 3 and 4 Technical Specification does not state RCS - Design pressure and Temperature.

2) Proposed Condition of License:

a. The revision is more complete than the current Technical Specification as follows:

RCS DESIGN PRESSURE and TEMPERATURES, and FSAR reference information are specified for consistency with the Standard Technical Specification.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards considerations are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed changes as described in item 2.a are similar to example (ii) of 48 FR 14870 in that they provide additional design information by including RCS DESIGN PRESSURE AND TEMPERATURES and FSAR reference.

Based on the above considerations the changes included in the development of proposed Technical Specification 5.4.1 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: RCS - VOLUME

NO: 5.4.2

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 5.2.3, REACTOR COOLANT SYSTEM.

2) Proposed Condition of License:

a. The amendment is more complete than the current Technical Specification as follows:

The total water and steam volume of the RCS at a specified temperature is provided for consistency with the Standard Technical Specification. The current Technical Specification provides only the liquid volume of the RCS and does not specify a corresponding temperature.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the technical specifications for example, a more stringent surveillance requirement.

1) The proposed changes as described in Item 2.a are similar to example (ii) of 48 FR 14870 in that they provide additional information by including the total RCS VOLUME and the corresponding RCS temperature.

Based on the above considerations the changes included in the development of proposed Technical Specification 5.4.2 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: RCS - SEISMIC STRESS

NO: 5.4.3

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in specification 5.2.2, Reactor Coolant System.

2) Proposed Condition of License:

a. The amendment reformats the design information contained in the current Technical Specification into the proposed specification.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards considerations are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature.

1) The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which reformats the design information contained in the current Technical Specification.

Based on the above considerations the changes included in the development of proposed Technical Specification 5.4.3 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION .

TITLE: METEOROLOGICAL TOWER LOCATION

NO: 5.5

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

The current Turkey Point Unit 3 and 4 Technical Specification does not identify the location of Meteorological Towers.

2) Proposed Condition of License:

a. The revision is more complete than the current Technical Specification as follows:

The locations of the Meteorological Towers are identified in the revised Specification Figure 5.1-1. Addition of this information is consistent with the Standard Technical Specification.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards considerations are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed change as described in Item 2.a is similar to example (ii) of 48 FR 14870 in that it provides additional information by including Meteorological Towers locations in the revised Technical Specification Figure 5.1-1.

Based on the above considerations the changes included in the development of proposed Technical Specification 5.5.1 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: FUEL STORAGE - CRITICALITY

NO: 5.6.1

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 5.4.2 and 5.4.3.

2) Proposed Condition of License:

a. The amendment reformats the design information contained in current Technical Specification into this specification for consistency with the Standard Technical Specification.

b. The revision is more complete than the current Technical Specification as follows:

1) The uncertainties allowances, $\Delta k/k$, have been specified for all spent fuel storage rack regions.

2) The nominal center-to-center distance between fuel assemblies have been specified for all spent fuel storage rack regions.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which consolidates current requirements into a technical specification format consistent with the Standard Technical Specifications and does not involve technical or plant modifications.

2) The proposed changes as described in Items 2.b.1 and 2.b.2 are similar to example (ii) of 48 FR 14870 in that they provide additional information by including uncertainties allowances and nominal center-to-center distances between fuel assemblies for all spent fuel storage rack regions.

Based on the above considerations the changes included in the development of proposed Technical Specification 5.6.1 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: FUEL STORAGE - DRAINAGE

NO: 5.6.2

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification does not include design information pertaining to inadvertent draining of storage pool.

2) Proposed Condition of License:

a. The revision is more complete than the current Technical Specification as follows:

Design features of the fuel storage pool pertaining to inadvertent drainage have been specified for consistency with the Standard Technical Specification.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed change as described in Item 2.a is similar to example (ii) of 48 FR 14870 in that it provides additional information by including design features of the fuel storage pool pertaining to inadvertent drainage.

Based on the above considerations the changes included in the development of proposed Technical Specification 5.6.2 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: FUEL STORAGE CAPACITY

NO: 5.6.3

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification does not include design information pertaining to FUEL STORAGE CAPACITY.

2) Proposed Condition of License:

a. The revision is more complete than the current Technical Specification as follows:

Design information on FUEL STORAGE CAPACITY has been specified for consistency with the Standard Technical Specification.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The changes in Item 2.a are similar to example (ii) of 48 FR 14870 in that it provides additional information by including the design capacity of the fuel storage pool.

Based on the above considerations the changes included in the development of proposed Technical Specification 5.6.3 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: FUEL STORAGE - STRUCTURE

NO: 5.6.4

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 5.4.1.

2) Proposed Condition of License:

a. The amendment reformats the FUEL STORAGE - STRUCTURE design features contained in current Technical Specification into the proposed specification.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature.

1) The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which reformats the FUEL STORAGE-STRUCTURE design features described in the current Technical Specification.

Based on the above considerations the changes included in the development of proposed Technical Specification 5.6.4 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: COMPONENT CYCLIC OR TRANSIENT LIMIT

NO: 5.7

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

The current Turkey Point Unit 3 and 4 Technical Specification does not contain information on COMPONENT CYCLIC OR TRANSIENT LIMIT.

2) The revision is more complete then the current Technical Specification as follows:

a. The COMPONENT CYCLIC OR TRANSIENT LIMIT design features are included in the proposed Technical Specification for consistency with the Standard Technical Specification.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed changes as described in Item 2.a are similar to example (ii) of 48 FR 14870 in that they provide additional design features by including COMPONENT CYCLIC OR TRANSIENT LIMIT information.

Based on the above considerations the changes included in the development of proposed Technical Specification 5.7.1 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: RESPONSIBILITY

NO: 6.1

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 6.1.1.

2) Proposed Condition of License:

a. The revision is more complete than the current Technical Specification as follows:

The amendment adds a requirement that a management directive be issued annually by the Group Vice President identifying individuals who will be responsible for control room command function. The addition of this requirement is consistent with the Standard Technical Specification.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed change as described in Item 2.a is similar to example (ii) of 48 FR 14870 in that it provides additional administrative controls by including a requirement that a management directive be issued annually by the Group Vice President identifying individuals who will be responsible for control room command function.

Based on the above considerations the changes included in the development of proposed Technical Specification 6.1 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: ORGANIZATION - OFFSITE

NO: 6.2.1

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 6.2.1.

2) Proposed Condition of License:

a. The amendment revises Figure 6.2-1 identifying Offsite Organization for facility management and technical support. The revised figure adds organizational detail of management and technical support for the plant. Addition of this information is consistent with the Standard Technical Specification.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature.

1) The proposed changes as described in Item 2.a are similar to example (i) of 48 FR 14870 in that they are administrative changes which consolidate current requirements into a technical specification format consistent with the Standard Technical Specifications and do not involve technical or plant modifications.

Based on the above considerations the changes included in the development of proposed Technical Specification 6.2.1 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: ORGANIZATION - FACILITY STAFF

NO: 6.2.2

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 6.2.2.

2) Proposed Condition of License:

a. The amendment reformats the current Technical Specification requirements for consistency with the Standard Technical Specification.

b. The revision is more complete than the current Technical Specification as follows:

The development of administrative procedures that limit the working hours of unit staff who perform safety-related function is specified for consistency with the Standard Technical Specification.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed changes as described in Item 2.a are similar to example (i) of 48 FR 14870 in that they are administrative changes which consolidate current requirements into a technical specification format consistent with the Standard Technical Specifications and do not involve technical or plant modifications.

2) The proposed changes as described in Item 2.b are similar to example (ii) of 48 FR 14870 in that they provide additional administrative controls by specifying development of procedures to limit the working hours of unit staff who perform safety-related functions.

Based on the above considerations the changes included in the development of proposed Technical Specification 6.2.2 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: SHIFT TECHNICAL ADVISOR

NO: 6.2.3

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described the current Turkey Point Unit 3 and 4 Technical Specification in Specification 6.3.1.

2) Proposed Condition of License:

a. The amendment reformats the current Technical Specification requirements for SHIFT TECHNICAL ADVISOR for consistency with the Standard Technical Specification.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature.

1) The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which reformats the requirements for the SHIFT TECHNICAL ADVISOR described in the current Technical Specification.

Based on the above considerations the changes included in the development of proposed Technical Specification 6.2.3 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: FACILITY STAFF QUALIFICATIONS

NO: 6.3

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 6.3.

2) Proposed Condition of License:

a. The amendment reformats the current Technical Specification requirements for Facility Staff. The proposed revision requires that the Health Physics Supervisor meet or exceed requirements of RG 1.8, September 1975 which are equal to the current requirements for this position.

b. The revision is more complete than the current Technical Specification as follows:

The licensed Operators and Senior Operators shall meet or exceed the minimum qualifications of the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees to the extent stated in Florida Power and Light Company response dated August 1, 1980.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed changes as described in Item 2.a are similar to example (i) of 48 FR 14870 in that they are administrative changes which consolidate current requirements into a technical specification format consistent with the Standard Technical Specification.

2) The proposed changes as described in Item 2.b are similar to example (ii) of 48 FR 14870 in that they provide additional restrictions by increasing licensed operators and senior operators qualification requirements.

Based on the above considerations the changes included in the development of proposed Technical Specification 6.3 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: TRAINING

NO: 6.4

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 6.4.

2) Proposed Condition of License:

a. The amendment consolidates the current requirements into the proposed specification for consistency with the Standard Technical Specification.

b. The revision is more complete than the current Technical Specification as follows:

The TRAINING program shall include familiarization with relevant industry operational experience. This requirement is consistent with the Standard Technical Specification.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which consolidates current requirements into a technical specification format consistent with the Standard Technical Specification.

2) The proposed change as described in Item 2.b is similar to example (ii) of 48 FR 14870 in that it provides an additional requirement to familiarize facility staff with relevant industry operational experience.

Based on the above considerations the changes included in the development of proposed Technical Specification 6.4 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: PLANT NUCLEAR SAFETY COMMITTEE

NO: 6.5.1

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 6.5.1.

2) Proposed Condition of License:

a. The amendment reforemats the functions of the PLANT NUCLEAR SAFETY COMMITTEE described in the current Technical Specification for consistency with Standard Technical Specification.

b. The revision is more complete than the current Technical Specification as follows:

The review of any accidental, unplanned, or uncontrolled radioactive release including the preparation of reports is specified as an additional responsibility of the PLANT NUCLEAR SAFETY COMMITTEE.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which reformats the PLANT NUCLEAR SAFETY COMMITTEE functions described in the current Technical Specification.

2) The proposed change as described in Item 2.b is similar to example (ii) of 48 FR 14870 in that it assigns an additional responsibility to the PLANT NUCLEAR SAFETY COMMITTEE.

Based on the above considerations the changes included in the development of proposed Technical Specification 6.5.1 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: COMPANY NUCLEAR REVIEW BOARD (CNRB)

NO: 6.5.2

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 6.5.2.

2) Proposed Condition of License:

a. The amendment reformats the current requirements for consistency with the Standard Technical Specification.

b. The revision is more complete than the current Technical Specification as follows:

1) The composition of the CNRB is updated.

2) CNRB audit frequency of EMERGENCY AND SECURITY Plans and their procedure implementation has changed to at least once per year from at least once per two years.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which reformats the CNRB functions described in the current Technical Specification.

2) The proposed changes as described in Items 2.b.1 and 2.b.2 are similar to example (ii) of 48 FR 14870 in that they add additional controls by increasing the membership of the CNRB and audit frequency of the Emergency and Security Plans and their procedure implementation.

Based on the above considerations the changes included in the development of proposed Technical Specification 6.5.2 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: REPORTABLE EVENT ACTION

NO: 6.6

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 6.6.

2) Proposed Condition of License:

a. The amendment replaces the current Reportable Occurence Action Section with Reportable Event Action. The revision requires compliance to requirements of Section 50.73 to 10 CFR Part 50. The revision is consistent with Standard Technical Specification.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature.

1) The proposed changes as described in Item 2.a are similar to example (i) of 48 FR 14870 in that they are administrative changes which consolidate current requirements into a technical specification format consistent with the Standard Technical Specifications.

Based on the above considerations the changes included in the development of proposed Technical Specification 6.6 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: SAFETY LIMIT VIOLATION

NO: 6.7

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

   As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 6.7.

2) Proposed Condition of License:

   a. The amendment reformats the current Technical Specification. The existing Technical Specification requirement to comply with 10 CFR 50.36 (C)(1)(i) has been replaced by explicit criteria in the text of Section 6.7. This change is considered administrative.

   b. The revision is more complete than the current Technical Specification as follows:

      1. The proposed revision specifically requires NRC notification by Emergency Notification System within 1 hour of a safety limit violation; this revision is consistent with the Standard Technical Specification. The existing Technical Specification requires that the NRC be informed immediately.

      2. The proposed revision requires that operation of the unit following a SAFETY LIMIT VIOLATION shall not be resumed until authorized by the Commission. This requirement is consistent with the Standard Technical Specification.

      3. The proposed revision requires the Group Vice President of Nuclear Energy be notified of a SAFETY LIMIT VIOLATION and receive the Safety Limit Violation Report.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which consolidates current requirements into a technical specification format consistent with the Standard Technical Specifications.

2) The proposed change as described in Item 2.b.1 is similar to example (ii) of 48 FR 14870 in that it provides an additional control by including a specific time interval for NRC notification.

3) The proposed change as described in Item 2.b.2 is similar to example (ii) of 48 FR 14870 in that it provides an additional control by requiring Commission's authorization prior to unit operation following a Safety Limit Violation.

4) The proposed change as described in Item 2.b.3 is similar to example (ii) of 48 FR 14870 in that it provides additional controls by requiring the Group Vice President of Nuclear Energy notification and review of a Safety Limit Violation Report.

Based on the above considerations the changes included in the development of proposed Technical Specification 6.7 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: PROCEDURES AND PROGRAMS

NO: 6.8

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 6.8, 6.13, 6.14, 6.15, and 6.16.

2) Proposed Condition of License:

a. The amendment reformats the current Technical Specification requirements for consistency with the Standard Technical Specification.

b. The revision is more complete than the current Technical Specification as follows:

1. Requirements to establish, implement and maintain written procedures and administrative policies for Emergency Operating Procedures, Security and Emergency Plans have been included in the proposed specification for consistency with the Standard Technical Specification.

2. Requirement to establish a program for monitoring Secondary Water Chemistry has been included in the proposed specification for consistency with the Standard Technical Specification.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed changes as described in Item 2.a are similar to example (i) of 48 FR 14870 in that they are administrative changes which consolidate current requirements into a technical specification format consistent with the Standard Technical Specifications.

2) The proposed change as described in Item 2.b.1 is similar to example (ii) of 48 FR 14870 in that it provides additional controls by including requirements to establish, implement and maintain written procedures and administrative policies for Emergency Operating Procedures, Security and Emergency Plans.

3) The proposed change as described in Item 2.b.2 is similar to example (ii) of 48 FR 14870 in that it provides additional controls by including requirements to establish a program for monitoring Secondary Water Chemistry.

Based on the above considerations the changes included in the development of proposed Technical Specification 6.8 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: REPORTING REQUIREMENTS - ROUTINE REPORTS

NO: 6.9.1

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specifications 6.9, 6.9.1, 6.9.3.d through 1, and 6.9.4 and Proposed Licensing Amendment -047 in proposed Specification 6.9.3. (FPL-NRC transmittal letter L-86-43, dated Jan. 30, 1986.)

2) Proposed Condition of License:

a. The amendment consolidates the current Technical Specification requirements into the proposed specification for consistency with the Standard Technical Specification format.

The reporting requirements for release of radioactive materials in liquid or gaseous effluents, steam generator tube inspections, power tilt ratio, standby feedwater and shutdown margin have been placed in their individual proposed technical specifications.

The reporting requirement for submittal of the Peaking Factor Limit Report 60 days prior to cycle initial criticality is replaced by 30 days. This change is considered administrative in nature, because the fuel vendor has determined that the Peaking Factor Limit Report is not generally available 60 days prior to initial cycle criticality and the proposed 30 days requirement is consistent with other Westinghouse plants.

b. The revision is more complete than the current Technical Specification as follows:

A reporting requirement for the results of specific activity in the primary coolant has been added. This requirement is consistent with the Standard Technical Specification.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed changes as described in Item 2.a are similar to example (i) of 48 FR 14870 in that they are administrative changes which consolidate current requirements into a technical specification format consistent with the Standard Technical Specifications.

2) The proposed change as described in Item 2.b is similar to example (ii) of 48 FR 14870 in that it provides an additional reporting requirement for the results of specific activity in the primary coolant.

Based on the above considerations the changes included in the development of proposed Technical Specification 6.9.1 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: SPECIAL REPORTING REQUIREMENTS

NO: 6.9.2

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 6.9.3 a through c.

2) Proposed Condition of License:

a. The amendment reformats the current Technical Specification reporting requirements into the proposed specification for consistency with Standard Technical Specification. This specification is referred to by individual proposed technical specifications for special report submittal when the individual specification conditions are not met.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature.

1) The proposed changes as described in Item 2.a are similar to example (i) of 48 FR 14870 in that they are administrative changes which consolidate current requirements into a technical specification format consistent with the Standard Technical Specifications.

Based on the above considerations the changes included in the development of proposed Technical Specification 6.9.2 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: RECORD RETENTION

NO: 6.10

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 6.10.

2) Proposed Condition of License:

a. The amendment reformats the current Technical Specification requirement into the proposed specification for consistency with the Standard Technical Specification. The existing requirement of retaining records for Environment Qualification has been deleted. This requirement should have been deleted by amendments 110/104 that replaced the Environmental Qualification Section 6.13 with Post Accident Monitoring. Retention of these records is not required by the Standard Technical Specification.

b. The revision is more complete than the current Technical Specification as follows:

1. Records of reactor tests and experiments are to be retained for the duration of the facility operating license in lieu of existing requirement to retain them for at least 5 years, and

2. An additional requirement is proposed that requires the records of secondary water sampling and water quality are to be retained for the duration of the facility operating license.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed changes as described in Item 2.a are similar to example (i) of 48 FR 14870 in that they are administrative changes which consolidate current requirements into a technical specification format consistent with the Standard Technical Specifications.

2) The proposed change as described in Item 2.b.1 is similar to example (ii) of 48 FR 14870 in that it provides additional control by requiring records of reactor tests and experiments retained for the duration of the facility operating license, in lieu of the existing 5 year requirement.

3) The proposed change as described in Item 2.b.2 is similar to example (ii) of 48 FR 14870 in that it provides additional information by requiring records of secondary water sampling and quality retained for the duration of the facility operating license.

Based on the above considerations the changes included in the development of proposed Technical Specification 6.10 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: RADIATION PROTECTION PROGRAM

NO: 6.11

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 6.11.

2) Proposed Condition of License:

a. The current Technical Specification is consistent with the Standard Technical Specification and therefore only minor reformatting changes are required.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature.

1) The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which reformats the current Technical Specification consistent with the Standard Technical Specification.

Based on the above considerations the changes included in the development of proposed Technical Specification 6.11 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: HIGH RADIATION AREA

NO: 6.12

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 6.12.

2) Proposed Condition of License:

a. The amendment reformats the current Technical Specification's HIGH RADIATION AREA requirements into the proposed technical specification for consistency with the Standard Technical Specification.

b. The revision is more complete than the current Technical Specification as follows:

1. The high radiation intensity is specified to be measured at 18 inches from the radiation source. The current Technical Specification does not specify a distance at which radiation levels should be measured.

2. The revision specifies HIGH RADIATION AREAS that are located within large areas, such as inside the containment, where no enclosure can be reasonably constructed around the individual shall be roped off, conspicuously posted and flashing lights activated as warning devices.

3. The revision specifies that Health Physics personnel and personnel escorted by them are allowed to enter a HIGH RADIATION AREA without a Radiation Work Permit (RWP) provided they are following Plant radiation protection procedures.

NOTE: The above additions are consistent with the Standard Technical Specifications.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed changes as described in Item 2.a are similar to example (i) of 48 FR 14870 in that they are administrative changes which consolidate current requirements into a technical specification format consistent with the Standard Technical Specifications.

2) The proposed change as described in Item 2.b.1 is similar to example (ii) of 48 FR 14870. This change provides an additional requirement by specifying that radiation intensities be measured 18 inches from the radiation source.

3) The proposed changes as described in Item 2.b.2 are similar to example (ii) of 48 FR 14870 in that they provide additional requirements by specifying an identification program for large hard to isolate, HIGH RADIATION AREAS.

4) The proposed change as described in Item 2.b.3 is similar to example (ii) of 48 FR 14870 in that it provides an additional control by specifying detailed HIGH RADIATION AREA access requirements.

Based on the above considerations the changes included in the development of proposed Technical Specification 6.12 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: PROCESS CONTROL PROGRAM (PCP)

NO: 6.13

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 6.17.

2) Proposed Condition of License:

a. The amendment reformats the current Technical Specification into this specification for consistency with Standard Technical Specification.

b. The revision is more complete than the current Technical Specification as follows:

The revision requires PCP approval by the Commission prior to implementation. The current Technical Specification requires that the PCP be reviewed by PNSC prior to implementation. This requirement is consistent with the Standard Technical Specifications.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout

the Technical Specifications, correction of an error, or a change in nomenclature. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed change as described in Item 2.a is similar to example (i) of 48 FR 14870 in that it is an administrative change which consolidates current requirements into a technical specification format consistent with the Standard Technical Specifications.

2) The proposed change as described in Item 2.b is similar to example (ii) of 48 FR 14870 in that it provides additional administrative control by requiring the PCP be approved by the Commission prior to implementation.

Based on the above considerations the changes included in the development of proposed Technical Specification 6.13 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: OFFSITE DOSE CALCULATION MANUAL (ODCM)

NO: 6.14

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

   As described in the current Turkey Point Unit 3 and 4 Technical Specification in Specification 6.18.

2) Proposed Condition of License:

   a. The revision reformats the current Technical Specification requirements for consistency with Standard Technical Specification. The revision deletes current Specification 6.18.1 that requires the ODCM be reviewed by the PNSC prior to the Commission submittal. Deletion of this is administrative in nature because this one time requirement has already been completed. In addition, any changes to the ODCM are required to be reviewed by the PNSC prior to submittal to the Commission per proposed Technical Specification 6.14.2.b.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, then a no significant hazards determination can be made.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature.

1) The proposed changes as described in Item 2.a are similar to example (i) of 48 FR 14870 in that they are administrative changes which consolidate current requirements into a technical specification format consistent with the Standard Technical Specifications.

Based on the above considerations the changes included in the development of proposed Technical Specification 6.14 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

NO SIGNIFICANT HAZARDS EVALUATION

PROPOSED TECHNICAL SPECIFICATION

TITLE: MAJOR CHANGES TO LIQUID, GASEOUS AND SOLID RADWASTE TREATMENT SYSTEMS

NO: 6.15

A. DESCRIPTION OF CHANGES

1) Present Condition of License:

The current Turkey Point Unit 3 and 4 Technical Specifications does not contain this specification.

2) Proposed Condition of License:

a. The revision is more complete than the current Technical Specifications as follows:

The specification requires that licensee initiated major changes to the Radwaste Treatment Systems be reported to the Commission in the Semiannual Radioactive Effluent Release Report. This proposed specification is consistent with the Standard Technical Specification.

B. BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION:

The standards used to arrive at a proposed determination that the changes described above involve no significant hazards consideration are included in 10 CFR 50.92. The regulations state that if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications for example, a more stringent surveillance requirement.

1) The proposed change as described in Item 2.a is similar to example (ii) of 48 FR 14870 in that it provides additional reporting requirements for major changes to the Radwaste Treatment Systems.

Based on the above considerations the changes included in the development of proposed Technical Specification 6.15 are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92. Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

ATTACHMENT III

TURKEY POINT UNITS 3 AND 4

CURRENT AND REVISED TECHNICAL SPECIFICATIONS

CROSS REFERENCE

September 30, 1986

III CURRENT/REVISED TECHNICAL SPECIFICATION CROSS REFERENCE

	Current PTP Technical Specifications		TSR
1.0	Definitions	1.0	Definitions
2.0	Safety Limits and LSSS	2.0	Safety Limits and LSSS
2.1	Rx Core	2.1.1	Rx Core
2.2	RCS Pressure	2.1.2	RCS Pressure
2.3	Rx Trip Setpoints	2.2.1	Rx Trip Setpoints
3.0.1	Motherhood	3.0.3	Motherhood
3.0.2	Operability	3.0.2	Operability
3.0.3	Compliance with LCO	3.0.1	Compliance with LCO
3.0.4	Operational Mode	3.0.4	Operational Mode
3.0.5	Component Operability with Redundant Power Supplies	3.0.5	Component Operability with Redundant Power Supplies
4.0.1	Grace Periods	4.0.2	Grace Period
4.0.2	Surveillance Performance	4.0.4	Mode Changes

III CURRENT/REVISED TECHNICAL SPECIFICATION CROSS REFERENCE

Section 3.1 to 3.16 (LCOs)

Current PTP Technical Specifications		TSR	
3.1.1.a.1	RCP	3.4.1.1	RCP at Power
		3.10.4	Special Test Exceptions
3.1.1.a.2	RCP	3.4.1.2	Hot Standby
		3.4.1.3	Hot Shutdown
		3.4.1.4.1	Cold Shutdown-Loops Filled
		3.4.1.4.2	Cold Shutdown
3.1.1.a.3	RCP 1 Loop Op.	3.4.1.1	RCP at Power
3.1.1.a.4	RCP 2 Loop Op.	3.4.1.1	RCP at Power
3.1.1.a.5	RCP Start Criteria	3.4.1.3	Hot Shutdown
		3.4.1.4.1	Cold Shutdown
3.1.1.b	S/G	3.4.5	S/G
3.1.1.c.1	PZR Safeties	3.4.2.1	PZR Safeties - Shutdown
3.1.1.c.2	PZR Safeties	3.4.2.2	PZR Safeties - Operating
3.1.1.d	PZR	3.4.3	PZR
3.1.1.e.1	PORV	3.4.4	PORV
3.1.1.e.2	PORV	3.4.4	PORV
3.1.1.e.3	PORV	3.4.4	PORV
3.1.1.f.1	RCS Vents	3.4.11	RCS Vent
3.1.1.f.2	RCS Vents	3.4.11	RCS Vent
3.1.1.f.3	RCS Vents	3.4.11	RCS Vent
3.1.1.f.4	RCS Vents	3.4.11	RCS Vent
3.1.2	Press/Temp. Limits	3.4.9.1	RCS Press/Temp Limits
		3.4.9.2	PZR Heat Up Cool Down
		3.4.10.3	Sprecial Tests
3.1.2.1	MTC	3.1.1.3	MTC
		3.10.3	Special Tests
3.1.3.a	Leakage	3.4.6.2	RCS Leakage
3.1.3.b	Leakage	3.4.6.2	RCS Leakage
3.1.3.c	Leakage	3.4.6.2	RCS Leakage
3.1.3.d	Safety Evaluation	(Note 1)	
3.1.3.e	Takle Corrective Action	3.0.4	Power Escalation
3.1.3.f	Leak Detection	3.4.6.1	Leak Detection
3.1.3.f	Leak Detection	3.3.3.1	Radiation Monitors
3.1.3.g	S/G Tube Leak	3.4.6.2	RCS Leakage
3.1.4.1	RCS Activity	3.4.8	RCS Activity
3.1.4.2	RCS Activity	3.4.8	RCS Activity
3.1.5.a	RCS Chemistry	3.4.7	RCS Chemistry
3.1.5.b	Corrective Action	3.4.7	RCS Chemistry
3.1.5.c	Transient Limits	3.4.7	RCS Chemistry
3.1.5.d	<250 F	3.4.7	RCS Chemistry
3.1.5.e	RCP Operation	(Note 2)	RCS Chemistry
3.1.6	DNB	3.2.5	DNB Parameters

III CURRENT/REVISED TECHNICAL SPECIFICATION CROSS REFERENCE

Current PTP Technical Specifications		TSR	
3.2.1.a	S/D Banks	3.1.3.5	S/D Banks
		3.10.2	Special Test
		3.10.3	Special Test
3.2.1.b	RIL	3.1.3.6	RIL
		3.10.2	Special Tests
		3.10.3	Special Tests
3.2.1.c	RIC	3.1.3.6	RIL
		3.10.2	Special Tests
		3.10.3	Special Tests
3.2.1.d	RIL Revision	(Note 3)	
3.2.1.e	Part Length	(Note 4)	
3.2.1.f	SDM	3.1.1.1	SDM
		3.1.1.2	SDM
		3.10.1	Special Test
		3.10.3	Special Test
3.2.1.g	Physics Test	3.10.2	Special Test
3.2.2	Misaligned Rod	3.1.3.1	Movable Control Rods
3.2.3	Rod Drop Time	3.1.3.4	Rod Drop Time
3.2.4.a	Inoperable Rod	3.1.3.1	Movable Control Rods
3.2.4.b	Inoperable Rod	3.1.3.1	Movable Control Rods
		3.1.3.4	Rod Drop Time
3.2.4.c	Immovable Rod	3.1.1.1	Boration Control (Interim T.S. Only)
		3.1.1.2	Boration Control (Interim T.S. Only).
3.2.5	Rod Position Indication	4.1.3.1.1	Movable Control Assemblies
		4.1.3.2.1	Position Indication System
3.2.6.a.1	HCF	3.2.2	Fq(z)
		3.2.3	$F^n_{\Delta h}$
3.2.6.a.2	Augment Surveillance	4.2.2.2	RCS ISI
3.2.6.a.3	Base Load	4.2.2.3	RCS ISI
3.2.6.a.4	Radial Burnup	4.2.2.4	RCS ISI
3.2.6.b	Fq(z), $F^n_{\Delta h}$	3.2.2	Fq(z)
		3.2.3	$F^n_{\Delta h}$
3.2.6.c	AFD	3.2.1	AFD
3.2.6.d	Target Bank ± 5 %	3.2.1	AFD
		3.10.2	Special Tests
3.2.6.e	Target Bank > 90%	3.2.1	AFD
3.2.6.f	AFD 50% to 90%	3.2.1	AFD
3.2.6.g	AFD <50%	3.2.1	AFD
3.2.6.h	QPTR	3.2.4	QPTR
		3.10.2	Special Tests

III CURRENT/REVISED TECHNICAL SPECIFICATION CROSS REFERENCE

Current PTP Technical Specifications		TSR	
3.2.6.i	QPTR	3.2.4	QPTR
3.2.7.a	Incore Inst.	3.3.3.2	MIDS
3.2.7.b	Incore Inst.	3.3.3.2	MIDS
3.2.8	AFD Alarms	4.2.1.1	AFD
3.3.1	CTMT Integrity	3.6.1.1	CTMT Integrity
3.3.2	CTMT Pressure	3.6.1.4	CTMT Pressure
3.3.3	CTMT Isolation	3.6.4	CTMT Isolation
3.4.1.a.1	RWST	3.5.4	RWST
		3.5.2	ECCS Subsystem
		3.1.2.6	Borated Water Source
		3.1.2.1	Boron Flow Path
		3.1.2.2	Boron Flow Path
3.4.1.a.2	BIT	(Note 6)	
3.4.1.a.3	Accumulators	3.5.1	Accumulators
3.4.1.a.4	SI Pumps	3.5.2	ECCS Subsystem
3.4.1.a.5	RHR	3.5.2	ECCS Subsystem
3.4.1.a.6	RHR Hx	3.5.2	ECCS Subsystem
3.4.1.a.7	Associated Components	3.5.2	ECCS Subsystem
3.4.1.b.1	Power Ops.	3.5.1	Accumulators
3.4.1.b.2	Power Ops.	3.5.2	ECCS Subsystem
3.4.1.b.3	Power Ops.	(Note 6)	
3.4.1.b.4	Power Ops.	3.5.2	ECCS
3.4.1.b.5	Power Ops.	3.5.2	ECCS
3.4.1.b.6	Power Ops.	3.5.2	ECCS
3.4.1.b.7	Valve Stroke	3.5.2	ECCS
3.4.1.c	Reactor Coolant Loops	3.4.1.1	Power Operation
3.4.1.d	Reactor Coolant Loops	3.4.1.2	Hot Standby
3.4.1.e	Reactor Coolant Loops	3.4.1.3	Hot Shutdown
3.4.2.a.1	CTMT Coolers	3.6.2.2	CTMT Coolers
3.4.2.a.2	CTMT Spray	3.6.2.1	CTMT Spray
3.4.2.a.3	Associated Components	3.6.2.1	CTMT Coolers
		3.6.2.2	CTMT Coolers
3.4.2.b.1	Power Ops	3.6.2.2	CTMT Coolers
3.4.2.b.2	Power Ops	3.6.2.1	CTMT Spray
3.4.2.b.3	Power Ops	3.6.2.1	CTMT Spray
		3.6.2.2	CTMT Coolers
3.4.3.a	CTMT Filters	3.6.3	CTMT Filtering Sys.
3.4.3.b	Associated Components	3.6.3	CTMT Filtering Sys.
3.4.4.a	CCW	3.7.3	CCW
3.4.4.b	Power Ops	3.7.3	CCW
3.4.5.a	Intake Water	3.7.4	Intake Water
3.4.5.b	Power Ops	3.7.4	Intake Water
3.4.6.a	Post Accident Vent.	3.6.6	Post Accident Vent.
3.4.6.b	Power Ops	3.6.6	Post Accident Vent.

III CURRENT/REVISED TECHNICAL SPECIFICATION CROSS REFERENCE

Current	PTP Technical Specifications		TSR
3.4.7.a	Control Rm Vent.	3.7.7	Control Rm. Vent.
3.4.7.b	Power Ops	3.7.7	Control Rm. Vent.
3.5.1	Table, Reactor Trip Instruments	3.3.1	RPS
3.5.2	Table, ESF Actuation	3.3.2	ESF Actuation
3.5.3	Table, Isolation Instruments	3.3.2	ESF Actuation
3.5.4	Table, ESF Setpoints	3.3.2	ESF Actuation
3.5.5	Table, Accident Monitoring Instruments	3.3.3.5	Post Accident Monitors
		3.6.5	Combustible Gas Control Monitors
3.6.a	Boron Flow Path	3.1.2.1	Boron Flow Path
		3.1.2.2	Boron Flow Path
		3.1.2.3	Charging Pump - Shutdown
3.6.b(c).1	Chg. Pump	3.1.2.4	Chg. Pump
3.6.b(c).2	BAT Pump	3.1.2.8	BAT Pump (Interim T.S. Only)
3.6.b(c).3	BAT	3.1.2.6	Borated Water Source
3.6.b(c).4	Associated Components	3.1.2.2	Boron Flow Path
3.6.b(c).5	BAT Heat Tracing	3.1.2.7	BAT Heat Tracing
3.6.b(c).6	Primary Water Storage Tank	3.4.12	PWST (Interim T.S. only)
3.6.d.1	Chg. Pump	3.1.2.4	Chg. Pump
3.6.d.2	BAT Pump	3.1.2.2	Boron Flow Path (Final T.S.)
		3.1.2.8	BAT Pump (Interim T.S. only)
3.6.d.3	Heat Tracing	3.1.2.7	BAT Heat Tracing
3.7.1.a	S/U Trans.	3.8.1.1	A.C. Sources
		3.8.1.2	A.C. Sources
3.7.1.b	4160 Bus	3.8.3	Onsite Power
3.7.1.c	480V. L.C.	3.8.3	Onsite Power
3.7.1.d	D/G	3.8.1	A.C. Sources
3.7.1.e	Batteries	3.8.2.1	D.C. Sources
3.7.2.a	Power OPS	3.8.1.1	A.C. Sources
3.7.2.b	Power OPS	3.8.1.1	A.C. Sources
3.7.2.c	Power OPS	3.8.2.1	D.C. Sources
3.7.2.d	Power OPS	3.0.3	Motherhood
3.8.1.a	S/G Safeties	3.7.1.1	S/G Safeties
3.8.1.b	CST (Changed to 3.19, See Note 8)		
3.8.1.c	MSIV	3.7.1.5	MSIV
3.8.1.d	Operability	3.7.1.1	S/G Safeties
		3.7.1.5	MSIV
3.8.2	Secondary Activity	3.7.1.4.	Specific Activity

III CURRENT/REVISED TECHNICAL SPECIFICATION CROSS REFERENCE

	Current PTP Technical Specifications		TSR
3.8.3	Power OPS.	3.7.1.1	S/G Safeties
		3.7.1.3	CST
		3.7.1.4	Specific Activity
		3.7.1.5	MSIV
3.8.4	AFW (Changed to 3.18, See Note 8)		
3.8.5	AFW (Changed to 3.18, See Note 8)		
3.9.1.a	Liquid Effluents	3.11.1.1	Liquid Effluents Concentration
3.9.1.b	Dose	3.11.1.2	Dose
3.9.1.c	Liquid Radioactive Effluent Instrumentation	3.3.3.6	Liquid Radioactive Effluent Instrumentation
3.9.1.d	Radwaste Treatment	3.11.1.3	Radwaste Treatment
3.9.2.a	Gaseous Dose Rate	3.11.2.1	Gaseous Dose
3.9.2.b	Noble Gases	3.11.2.2	Noble Gases
3.9.2.c	Dose I-131	3.11.2.3	Dose I-131
3.9.2.d	Gaseous Radioactive Effluent Monitoring	3.3.3.7	Gaseous Radioactive Effluent Monitoring
3.9.2.e	Radwaste Treatment	3.11.2.4	Radwaste Treatment
3.9.2.f	Gas Decay Tanks	3.11.2.6	Gas Decay Tanks
3.9.2.g	Explosive Mixture	3.11.2.5	Explosive Mixture
3.9.2.h	Total Dose	3.11.4	Total Dose
3.9.3	Resin Beads	3.11.3	Solid Radioactive Wastes
3.10.1	CTMT Integrity	3.9.4	CTMT Integrity
3.10.2	Purge	3.3.3.1	Radiation Monitors
		3.9.9	CTMT Purge
3.10.3	Instrumentation	3.9.2	Instrumentation
3.10.4	Radiation Monitoring	3.9.13	Radiation Monitoring
3.10.5	Decay Time	3.9.3	Decay Time
3.10.6	Communications	3.9.5	Communications
3.10.7	RHR	3.9.8.1	RHR - High Water Level
		3.9.8.2	RHR - Low Water Level
3.10.8	Boron Concentration	3.9.1	Boron Concentration
3.10.9	Crane Travel - Spent Fuel Storage Areas	3.9.7	Crane Travel - Spent Fuel Storage Areas
3.11.1	Removable Contamination	3.7.9	Sealed Source Contamination
3.11.2	Material Inventory Radioactive	4.7.9	Sealed Source Contamination
3.12.1	Cask Handling	3.9.12	Cask Handling
3.12.2	Cask Handling	3.9.12	Cask Handling
3.12.3	Cask Decay Time	3.9.12	Cask Handling
3.12.4	Heavy Load	3.9.7	Crane Travel

III CURRENT/REVISED TECHNICAL SPECIFICATION CROSS REFERENCE

	Current PTP Technical Specifications		TSR
3.13.1	Snubbers	3.7.9	Snubbers
3.13.2	Snubbers	3.7.9	Snubbers
3.13.3	Snubbers	3.7.9	Snubbers
3.13.4	Add/Remove Snubbers	3.7.9	Snubbers
3.14.1.a	Fire Detection	3.3.3.6	Fire Detection
3.14.1.b	Fire Detection	3.3.3.6	Fire Detection
3.14.1.c	Turb Area Fire Watch	3.3.3.6	Fire Detection
3.14.2.a	Fire Suppression	3.7.10.1	Fire Supply System
3.14.2.b	Fire Suppression	3.7.10.1	Fire Supply System
3.14.2.c	Fire Suppression	3.7.10.1	Fire Supply System
3.14.3.a	Spray/Sprinklers	3.7.10.2	Spray/Sprinkler Systems
3.14.3.b	Spray/Sprinklers	3.7.10.2	Spray/Sprinkler Systems
3.14.4	Hoses	3.7.10.4	Hose Stations
3.14.5.a	Fire Barriers	3.7.11	Fire Barriers
3.14.5.b	Fire Barriers	3.7.11	Fire Barriers
3.15.1	RCS Press Boundary Integrity	3.4.9.3	Overpressure Protection
3.15.2	3.15.1 Action	3.4.9.3	Overpressure Protection
3.15.3.a	Overpressure/PORV	3.4.9.3	Overpressure protection
3.15.3.b	PORV	3.4.9.3	Overpressure Protection
3.16.1	RCS Pressure Isolation Valves	4.4.6.2	RCS Leakage
3.16.2.a	Leakage	3.4.6.2	RCS Leakage
3.16.2.a	Leakage	3.4.6.2	RCS Leakage
3.16.2.c	Leakage	3.4.6.2	RCS Leakage
3.16.3	Operability	3.4.6.2	RCS Leakage
3.16.4	Leakage	3.4.6.2	RCS Leakage
3.18	AFW	3.7.1.2	AFW (See Note 8)
3.19	CST	3.7.1.3	CST (See Note 8)
3.20	Standby F.W.	3.7.1.6	Standby F.W.(See Note 9)

III CURRENT/REVISED TECHNICAL SPECIFICATION CROSS REFERENCE

Section 4.0 Surveillance Requirements

	Current PTP Technical Specifications	TSR	
	TABLE 4.1-1		
1.a.	Power Range	4.3.1	RPS
b.	Flux Map	4.2.2	$F_Q(Z)$
		4.2.3	$F^N_{\Delta H}$
2.	Intermediate Range	4.3.1	RPS
3.	Source Range	4.3.1	RPS
4.	OPΔT, OTΔT	4.3.1	RPS
5.	RCS flow	4.2.5	DNB
6.	PZR Level	4.3.1	RPS
7.	PZR Press	4.3.1	RPS
		4.3.2	ESF
8.	U/V and U/F	4.3.1	RPS
9.	Rod Position	4.1.3.2	Rod Position
		4.1.3.3	Position Indicating-System Shutdown
10.	Rod Position	4.1.3.2	Rod Position
		4.1.3.3	Position Indicating - System Shutdown
11.	S/G Level	4.3.1	RPS
12.	Chg. Flow	4.1.2.4	Charging Pumps
13.	RHR Flow	4.5.2	ECCS
14.	BAT Level	4.1.2.5	Borated Water Source - Shutdown (Interim Tech. Spec. only)
		4.1.2.6	Borated Water Source - Operation (Interim Tech. Spec. only)
		4.3.3.5	Accident Monitors
15.	RWST	4.1.2.5	Borated Water Source - Shutdown (Interim Tech. Spec. only)
		4.1.2.6	Borated Water Source - Operation (Interim Tech. Spec. only)
		4.3.3.5	Accident Monitors
16.	VCT Level	4.1.2.4	Charging Pumps (Interim Tech. Spec. only)

III CURRENT/REVISED TECHNICAL SPECIFICATION CROSS REFERENCE

	Current PTP Technical Specifications		TSR
17a.	CTMT Press	4.3.2	ESF
17b.	CTMT Press	4.3.2	ESF
18a.	Process Radiation	3.3.3.1	Radiation Monitors
18b.	Area Radiation	3.3.3.1	Radiation Monitors
19.	Boric Acid Control	4.1.2.1	Boron Flow Path
20.	CTMT Sump	4.4.6.1	Leakage Detection System
21.	Accumulator Level and Pressure	4.5.1	Accumulator
22.	Steam Line Pressure	4.3.2	ESF (Table 3.3-2, Item 4d)
23.	Logic Channels	4.3.1	RPS
24.	Emerg. Portable Survey (See Note 11)	4.3.3.5	Accident Monitoring Instrumentation
25.	Seismograph	4.3.3.3	Seismic Instrumentation
26.	AFW Flow	4.3.3.5	Accident Monitors (Table 3.3-6, Item 5)
27.	Subcooling Monitor	4.3.3.5	Accident Monitors (Table 3.3-6, Item 6)
28.	PORV Indication	4.3.3.5	Accident Monitors (Table 3.3-6, Item 7)
29.	PORV Block Valve Indication	4.3.3.5	Accident Monitors (Table 3.3-6, Item 8)
30.	Safeties Indication	4.3.3.5	Accident Monitors (Table 3.3-6, Item 9)
31.	LOSP	4.3.2	ESF (Table 3.3-2, Item 7)
32.	Main Feed Pump Bkrs.	4.3.2	ESF (Table 3.3-2, Item 6e)
33.	CTMT Water Level (Narow Range)	4.3.3.5	Accident Monitors (Table 3.3-6, Item 13)
34.	CTMT Water Level (Wide Range)	4.3.3.5	Accident Monitors (Table 3.3-6, Item 12)
35.	CTMT High Range Radiation	4.3.3.5	Accident Monitors (Table 3.3-6, Item 14)
		4.3.3.5	Accident Monitors (Table 3.3-6, Item 15)
36.	CTMT Hydrogen	4.6.5	Combustible Gas Control Monitors
37.	Noble Gas Monitors	4.3.3.5	Accident Monitors (Table 3.3-6, Item 16)
38.	Incore Thermocouples	4.3.3.5	Accident Monitors (Table 3.3-6, Item 17)
	TABLE 4.1-2		
1.a.	RCS $T_{1/2}$ > 30 min.	4.4.8	RCS Activity
b.	CL.-, O_2 and F	4.4.7	RCS Chemistry
c.	H_3	4.4.8	RCS Activity

III CURRENT/REVISED TECHNICAL SPECIFICATION CROSS REFERENCE

	Current PTP Technical Specifications	TSR	
	TABLE 4.1-2 (Continued)		
d.	Gross Activity	4.4.8	RCS Activity
e.	Boron Concentration	4.1.1.1	SDM > 200 F
		4.1.1.2	SDM ≤ 200 F
f.	Ē	4.4.8	RCS Activity
g.	Isotopic Dose Equiv. I-131	4.4.8	RCS Acitvity
h.	Isotopic for I-131, I-133, I-135	4.4.8	RCS Activity
2.	RWST	4.3.3.5	Accident Monitoring (Table 3.3-6, Item 18)
		4.1.2.6	Boron Water Sources
		4.5.4	RWST
3.	Boric Acid Tank	4.1.2.5	Boron Flow Path
		4.1.2.6	Boron Flow Path
		4.3.3.5	Accident Monitors (Table 3.3-6, Item 19)
4.	BIT	(Note 6)	
5.	Control Rod Drop Time	4.1.3.4	Rod Drop Time
		4.1.3.1	Movable Rods
6.	PZR Safeties	4.4.3	Safeties
7.	S/G Safeties	4.7.1.1	S/G Safeties
8.	CTMT Isolation	4.3.2	ESF
9.	Refueling System	4.9.6	Manipulator Crane
10.	Accumulator	4.5.1	Accumulator
11.	RCS Leakage	4.4.6.2	RCS Leakage
12.	D/G Fuel	4.8.1	A.C. Sources
		4.8.2	A.C. Sources
13.	SFP	4.9.1	Boron Concentration
14.	Fire Protection Pump	4.7.10.1	Fire Water Supply System
15.	Turbine Stop and Control Valve	4.3.4	Turbine Overspeed
16.	L.P. Turbine Inspection	4.3.4	Turbine Overspeed
17.	Cask Crane	4.9.12	Cask Handling
18	Coolant Loops	4.4.1.1	Power Operation
		4.4.1.2	Hot Standby
		4.4.1.3	Hot Shutdown
		4.9.8.1	Refueling RHR
	TABLE 4.1-3 Liquid Effluents	4.3.3.6	Liquid Effluents
	TABLE 4.1-4 Gaseous Effluents	4.3.3.7	Gaseous Effluents
4.2.1.a	RCS Inspection	(Note 6)	
4.2.1.b	RCS Inspection	4.4.10	RCS Structural Integrity
4.2.1.c	RCS Inspection	3.4.10	RCS Structural Integrity
4.2.1.d	RCS Inspection	4.4.10	RCS Structural Integrity

III CURRENT/REVISED TECHNICAL SPECIFICATION CROSS REFERENCE

Current PTP Technical Specifications		TSR	
4.2.2	Structural Integrity	4.4.10	RCS Structural Integrity
4.2.3	Structural Integrity	4.4.10	RCS Structural Integrity
4.2.4	Structural Integrity	3.4.10	RCS Structural Integrity
4.2.5	S/G Inspection	4.4.5	S/G Inspection
4.3	RCS Pressure Boundary	4.4.10	RCS Integrity
4.4.1	LLRT	4.6.1.2	CTMT Leakage
		3.6.1.2	CTMT Leakage
4.4.2.1	LLRT	4.6.1.2	Leakage
		4.6.1.7	Purge
4.4.2.2	LLRT	4.6.1.2	Leakage
		4.6.1.3	Air Locks
4.4.2.3	LLRT	4.6.1.2	Type B LLRT
4.4.2.4	LLRT	4.6.1.2	Type B LLRT
4.4.2.5	LLRT	4.6.1.2	Type B LLRT
4.4.3	LLRT	4.6.1.2	Type C LLRT
4.4.4	RHR	(Deleted, See Note 10)	
4.4.5	Tendons	4.6.1.6	CTMT Structural Integrity
4.4.6	End Anchor	4.6.1.6	CTMT Structural Integrity
4.4.7	Liner	4.6.1.6	CTMT Structural Integrity
4.5.1	SI Test	4.5.2	ECCS
4.5.2.a	Pumps Operation	4.5.2	ECCS
4.5.2.b.1	Valve Cycle Test	4.5.2	ECCS
4.5.2.b.2	Valve Cycle Test	4.5.2	ECCS/Sump suction
4.5.2.b.3	Valve Cycle Test	4.5.1	Accumulators
4.5.2.b.4	Valve Cycle Test	4.5.2	ECCS
4.6.1	CTMT Cooling	4.6.2.2	CTMT Coolers
		4.6.2.1	CTMT Spray
4.6.2	Component Test	4.6.2.2	CTMT Coolers
		4.6.2.1	CTMT Spray
4.7.1	CTMT Filters	4.6.3	CTMT Filtering Sys.
4.7.2	Post Accident Vent	4.6.6	Post Accident CTMT Vent Sys.
4.7.3	Control Room Vent.	4.7.7	Control Room Vent
4.8.1	D/G	4.8.1	D/G
4.8.2	Batteries	4.8.2	Batteries
4.9	MSIV	4.7.1.5	MSIV
4.10	AFW	4.7.1.2	AFW
4.11	Reactivity Anomalies	4.1.1.1.2	SDM
4.12	Envir. Radiation Monitors	3/4.12.1	Envir. Radiation Monitors
4.13	Sealed Source	4.7.9	Sealed Source
4.14	Snubbers	4.7.8	Snubbers

III CURRENT/REVISED TECHNICAL SPECIFICATION CROSS REFERENCE

	Current PTP Technical Specifications		TSR
4.15.1	Fire Detection System	4.3.3.6	Fire Detection Instrumentation
4.15.2	Fire Protection Water System	4.7.10.1	Fire Water Supply System
4.15.3	Sprinkler	4.7.10.2	Fire Sprinklers
4.15.4	Hoses	4.7.10.4	Hoses
4.15.5	Fire Barriers	4.7.11	Fire Barriers
4.16.1	Overpressure	4.4.9.3.1	Overpressure
4.16.2	Overpressure	4.4.9.3.1	Overpressure
4.16.3	Overpressure	4.4.9.3.1	Overpressure
4.16.4	Overpressure	4.4.9.3.1	Overpressure
4.16.5	Backup Air Supply	4.4.9.3.1	Overpressure
4.16.6	Use Procedures	6.8.1	Approved Procedures
4.17.1	RCS Pressure Valves	4.4.6.2	Leakage
4.17.2	RCS Pressure Valves	3.4.6.2	RCS Leakage
4.18.1	HHSI Walkdown	4.5.2	ECCS
4.18.2	LHSI	4.5.3	ECCS
4.18.3	AFW	4.7.1.2	AFW
4.18.4	CTMT Spray	4.6.2.1	CTMT Spray
4.18.5	D/G	4.8.1	D/G
4.18.6	CCW	4.7.3	CCW
4.18.7	Intake Cooling	4.7.4	Intake Cooling Water
4.18.8	Boron Flow	4.1.2.1	Boron Flow Path
		4.1.2.2	Boron Flow Path
		4.1.2.7	BAT Heat Tracing
4.18.9	Post Accident Vent	4.6.6	CTMT Vent
4.18.10	In-Plant A.C.	4.8.2.1	A.C. Sources
		4.8.2.2	A.C. Sources
4.18.11	H_2 Monitor	4.6.5	Combustible Gas Control Monitors
4.18.12	Post Accident Samples	4.3.3.5	Post Accident Monitors
4.18.13	Fire Suppression	4.7.10.1	Fire Water Supply
4.21	Standby F.W. (New T.S., See Note 9)	4.7.1.6	Standby F.W.

III CURRENT/REVISED TECHNICAL SPECIFICATION CROSS REFERENCE

5.0 DESIGN FEATURES

Current PTP Technical Specifications		TSR	
5.1	Site	5.1.1	Exclusion Area
5.2.1	Reactor - Number of Rods	5.3.1	Fuel Assemblies
5.2.1	RCS Code Compliance	5.4.1	RCS - Design Pressure and Temperature
5.2.2	RCS - Seismic Design	5.4.3	RCS - Seismic Stress
5.2.3	RCS Volume	5.4.2	RCS - Volume
5.2.4	BPRA	5.3.3	BPRA
5.2.5	Control Rods	5.3.2	Control Rod Assemblies
5.3.A.1	CTMT	5.2.6	CTMT-Function
5.3.A.2	Design Pressure	5.2.2	CTMT-Design Pressure and Temperature
5.3.A.2	CTMT	5.2.3	CTMT-Structural Loads
5.3.B.1	Hatches and Electric Penetration	5.2.4.1	CTMT Design
5.3.B.2	CTMT Isolation Valve	5.2.4.2	CTMT Design
5.3.C.1	CTMT Coolers	5.2.5.1	CTMT Systems
5.3.C.2	CTMT Spray	5.2.5.2	CTMT Systems
5.3.C.3	CTMT Filters	5.2.5.3	CTMT Systems
5.4.1	SFP Structure	5.6.4	Fuel Storage Structure
5.4.2	Storage	5.6.1.1 5.6.1.2	Spent Fuel New Fuel
5.4.3	Burnup	5.6.1.4	Burnup

III CURRENT/REVISED TECHNICAL SPECIFICATION CROSS REFERENCE

Section 6.0 Administrative Control

Current PTP Technical Specifications		TSR	
6.1.1	Plant Manager	6.1.1	Plant Manager
6.2.1	Offsite	6.2.1	Offsite
6.2.2	Facility Staff	6.2.2	Facility Staff
6.3.1	Staff Qualification	6.3.1	Staff Qualification
		6.2.4	Shift Technical Advisor
6.3.2	H.P. Supv. Qualification	6.3.2	Staff Qualification
6.4.1	Training	6.4.1	Training
6.5.1	Review and Audit	6.5.1	Review and Audit
6.5.1.1	Function	6.5.1.1	Function
6.5.1.2	Composition	6.5.1.2	Composition
6.5.1.3	Alternates	6.5.1.3	Alternate
6.5.1.4	Meeting Frequency	6.5.1.4	Meeting Frequency
6.5.1.5	Quorum	6.5.1.5	Quorum
6.5.1.6	Responsibilities	6.5.1.6	Responsibilities
6.5.1.7	Authority	6.5.1.7	Authority
6.5.1.8	Records	6.5.1.8	Records
6.5.2	CNRB	6.5.2	Review Group
6.5.2.1	Function	6.5.2.1	Function
6.5.2.2	Composition	6.5.2.2	Composition
6.5.2.3	Alternate	6.5.2.3	Alternate
6.5.2.4	Consultants	6.5.2.4	Consultants
6.5.2.5	Meeting Frequency	6.5.2.5	Meeting Freq.
6.5.2.6	Quorum	6.5.2.6	Quorum
6.5.2.7	Review	6.5.2.7	Review
6.5.2.8	Audits	6.5.2.8	Audits
6.5.2.9	Authority	6.5.2.1	Function
6.5.2.10	Records	6.5.2.9	Records
6.6.1	Reportable Occurrence	6.6.1	Reportable Events
6.7.1	Safety Limit Violation	6.7.1	Safety Limit Violation
6.8.1	Procedures	6.8.1	Procedures
6.8.2	Approval	6.8.2	Approval
6.8.3	Temporary Changes	6.8.3	Temporary Changes
6.9.1	Routine Reports	6.9.1	Routine Reports
		6.9.1.1	Startup Report
		6.9.1.2	Annual Reports
6.9.4b	Annual Radiological Environmental Operating Report	6.9.1.3	Annual Radiological Environmental Operating Report
6.9.4a	Semiannual Radioactive Effluent Release Report	6.9.1.4	Semiannual Radioactive Effluent Release Report
6.9.1c	Monthly Operating Report	6.9.1.5	Monthly Operating Report
6.9.3a	Special Reports	6.9.2	Special Reporting Requirements

III CURRENT/REVISED TECHNICAL SPECIFICATION CROSS REFERENCE

Section 6.0 Administrative Control

Current PTP Technical Specifications		TSR	
6.9.3b	Special Reports	6.9.2	Special Reporting Requirements
6.9.3c	Special Reports	6.9.2	Special Reporting Requirements
6.9.3d	Peaking Factor Limit Report	6.9.1.6	Peaking Factor Report
6.9.3e	Special Reports	3/4.11.1.2	Liquid Effluent Dose
6.9.3e	Special Reports	3/4.11.2.2	Gaseous Effluent Dose
6.9.3g	Special Reports	3/4.11.2.4	Gaseous Radwaste Treatment System
6.9.3h	Special Reports	3/4.11.4	Total Dose
6.9.3i	Special Reports	3/4.12.1	Monitoring Program
6.9.3j	Special Reports	3/4.4.5	Steam Generators Quadrant Power (See Note 10)
		3/4.7.1.6	Standby Feedwater System (See Note 9)
6.9.3k	Special Reports	3/4.2.4	Tilt Ratio
6.9.3l	Special Reports	3/4.1.1.2	Shutdown Margin
6.10	Records Retention	6.10	Records Retention
6.10.1	5 Yr. Retention	6.10.2	5 Yr. Retention
6.10.2	Duration Retention	6.10.3	Duration Retention
6.11	Radiation Protection Program	6.11	Radiation Protection Program
6.12	High Radiation Areas	6.12	High Radiation Area
6.13	Post Accident	6.8.4e	Post Accident Sampling
6.14	System Integrity	6.8.4a	Primary Coolant
6.15	Iodine Monitor	6.8.4b	In-Plant Radiation
6.16	Backup Subcool Monitor	6.8.4.d	Subcooling Monitor
6.17	Process Control Program	6.13	Process Control Program
6.18	Offsite Dose Calculation Manual	6.14	Offsite Dose Calculation Manual

Note 1 STS does not address safety evaluation to determine if shutdown is required. Only leakage limits are used.

Note 2 STS does not state that RCP operation is not allowed. The STS applicability is greater than 200 F. Mod 4 is 200 F - 350 F and encompasses the plus 50 F allowance of PTP TS.

Note 3 RIL revision is not addressed by STS. The limits are set based on physics test performed.

Note 4 Part length rods specification no longer applicable.

Note 5 (deleted)

Note 6 These requirements are no longer applicable at PTP.

Note 7 (deleted)

Note 8 Per Proposed Licensing Amendment 039, transmittal letter No.L-86-193, dated May 7, 1986.

Note 9 Per Proposed Licensing Amendment 047, transmittal letter No.L-86-43, dated January 30, 1986.

Note 10 Per Proposed Licensing Amendment, transmittal letter No. L-85-346, dated October 11, 1986.

Note 11 This requirement has been deleted in the TSR (See applicable No Significant Hazards Evaluation).

ATTACHMENT I

TURKEY POINT UNITS 3 AND 4

REVISED TECHNICAL SPECIFICATIONS

September 30, 1986

8610140042

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION 1.0

DEFINITIONS

1.0 DEFINITIONS

The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications.

ACTION

1.1 ACTION shall be that part of a Technical Specification which prescribes remedial measures required under designated conditions.

ACTUATION LOGIC TEST

1.2 An ACTUATION LOGIC TEST shall be the application of various simulated input combinations in conjunction with each possible interlock logic state and verification of the required logic output. The ACTUATION LOGIC TEST shall include a continuity check, as a minimum, of output devices.

ANALOG CHANNEL OPERATIONAL TEST

1.3 An ANALOG CHANNEL OPERATIONAL TEST shall be the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY of alarm, interlock and/or trip functions. The ANALOG CHANNEL OPERATIONAL TEST shall include adjustments, as necessary, of the alarm, interlock and/or Trip Setpoints such that the Setpoints are within the required range and accuracy.

AXIAL FLUX DIFFERENCE

1.4 AXIAL FLUX DIFFERENCE shall be the difference in normalized flux signals between the top and bottom halves of a two section excore neutron detector.

CHANNEL CALIBRATION

1.5 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds within the required range and accuracy to known values of input. The CHANNEL CALIBRATION shall encompass the entire channel including the sensors and alarm, interlock and/or trip functions and may be performed by any series of sequential, overlapping, or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

1.6 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter, or radioactive SOURCE CHECK of the Area and Process Radiation Monitoring System channels.

CONTAINMENT INTEGRITY

1.7 CONTAINMENT INTEGRITY shall exist when:

a. All penetrations required to be closed during accident conditions are either:

1) Capable of being closed by an OPERABLE containment automatic isolation valve system, or

2) Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Table 3.6-1 of Specification 3.6.4.

b. The equipment hatch is closed and sealed,

c. Each air lock is in compliance with the requirements of Specification 3.6.1.3,

d. The containment leakage rates are within the limits of Specification 3.6.1.2.

CORE ALTERATION

1.8 CORE ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

DOSE EQUIVALENT I-131

1.9 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites" or Table E-7 of NRC Regulatory Guide 1.109, Revision 1, October 1977.

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.10 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (MeV/d) for the isotopes, other than iodines, with half-lives greater than 30 minutes, based upon those energy peaks identified with a 95% confidence level.

1.0 DEFINITIONS

FREQUENCY NOTATION

1.11 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

GAS DECAY TANK SYSTEM

1.12 A GAS DECAY TANK SYSTEM shall be any system designed and installed to reduce radioactive gaseous effluents by collecting Reactor Coolant System off gases from the Reactor Coolant System and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

IDENTIFIED LEAKAGE

1.13 IDENTIFIED LEAKAGE shall be:

a. Leakage (except for RCP seal leak-off) into closed systems, such as pump seal or valve packing leaks that are captured and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of Leakage Detection Systems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor Coolant System leakage through a steam generator to the Secondary Coolant System.

MASTER RELAY TEST

1.14 A MASTER RELAY TEST shall be the energization of each master relay and verification of OPERABILITY of each relay. The MASTER RELAY TEST shall include a continuity check of each associated testable slave relay.

MEMBER(S) OF THE PUBLIC

1.15 MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the licensee, its contractors, vendors or members of the Armed Forces using property located within the SITE BOUNDARY. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with the plant.

1.0 DEFINITIONS

OFFSITE DOSE CALCULATION MANUAL

1.16 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program.

OPERABLE - OPERABILITY

1.17 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s) and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its function(s) are also capable of performing their related support function(s).

OPERATIONAL MODE - MODE

1.18 An OPERATIONAL MODE (i.e., MODE) shall correspond to any one inclusive combination of core reactivity condition, power level, and average reactor coolant temperature specified in Table 1.2.

PHYSICS TESTS

1.19 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation: (1) described in ANSI 19.6.1, RELOAD STARTUP PHYSICS TESTS FOR PWR's, (2) authorized under the provisions of 10CFR50.59, or (3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

1.20 PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a nonisolable fault in a Reactor Coolant System component body, pipe wall, or vessel wall.

PROCESS CONTROL PROGRAM

1.21 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, tests, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71 and Federal and State regulations, burial ground requirements, and other requirements governing the disposal of radioactive waste.

1.0 DEFINITIONS

PURGE - PURGING

1.22 PURGE or PURGING shall be any controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

QUADRANT POWER TILT RATIO

1.23 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the averiage.

RATED THERMAL POWER

1.24 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 2200 MWt.

REFERENCE POSITION

1.25 Analog Rod Position Indication System REFERENCE POSITION is defined as:

a. For all Shutdown Banks and Control Banks A and B; the group demand counter indicated position between 0 and 30 steps withdrawn inclusive and between 200 and 228 steps withdrawn inclusive.

b. For Control Banks C and D; the group demand counter indicated position between 0 and 30 steps withdrawn inclusive and between 150 and 228 steps withdrawn inclusive. For the withdrawal range of 31 to 149 steps inclusive the REFERENCE POSITION shall be the individual rod calibration curve noting indicated analog rod position vs indicated group demand counter position.

REPORTABLE EVENT

1.26 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 of 10 CFR Part 50.

1.0 DEFINITIONS

SHUTDOWN MARGIN

1.27 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all full-length rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.

SITE BOUNDARY

1.28 The SITE BOUNDARY shall be that line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.

SLAVE RELAY TEST

1.29 A SLAVE RELAY TEST shall be the energization of each slave relay and verification of OPERABILITY of each relay. The SLAVE RELAY TEST shall include a continuity check, as a minimum, of associated testable actuation devices.

SOLIDIFICATION

1.30 SOLIDIFICATION shall be the conversion of wet wastes into a form that meets shipping and burial ground requirements.

SOURCE CHECK

1.31 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a source of increased radioactivity.

STAGGERED TEST BASIS

1.32 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains, or other designated components obtained by dividing the specified test interval into n equal subintervals, and

b. The testing of one system, subsystem, train, or other designated component at the beginning of each subinterval.

THERMAL POWER

1.33 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

TRIP ACTUATING DEVICE OPERATIONAL TEST

1.34 A TRIP ACTUATING DEVICE OPERATIONAL TEST shall consist of operating the Trip Actuating Device and verifying OPERABILITY of alarm, interlock and/or trip functions. The TRIP ACTUATING DEVICE OPERATIONAL TEST shall include adjustment, as necessary, of the Trip Actuating Device such that it actuates at the required Setpoint within the required accuracy.

UNIDENTIFIED LEAKAGE

1.35 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE or RCP seal leak-off.

UNRESTRICTED AREA

1.36 An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY access to which is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials, or any area within the SITE BOUNDARY used for residential quarters or for industrial, commercial, institutional, and/or recreational purposes.

VENTILATION EXHAUST TREATMENT SYSTEM

1.37 A VENTILATION EXHAUST TREATMENT SYSTEM shall be any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment. Such a system is not considered to have any effect on noble gas effluents. Engineered Safety Features atmospheric cleanup systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

VENTING

1.38 VENTING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration, or other operating condition, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.

TABLE 1.1

FREQUENCY NOTATION

NOTATION	FREQUENCY
S	At least once per 12 hours
D	At least once per 24 hours
W	At least once per 7 days
4/M	At least 4 per month at intervals of no greater than 9 days and a minimum of 48 per year
M	At least once per 31 days
Q	At least once per 92 days
SA	At least once per 184 days
R	At least once per refueling
S/U	Prior to each reactor startup
N.A.	Not applicable
P	Completed prior to each release

TABLE 1.2

OPERATIONAL MODES

MODE	REACTIVITY CONDITION, K_{eff}	% RATED THERMAL POWER*	AVERAGE COOLANT TEMPERATURE
1. POWER OPERATION	$\geq$ 0.99	> 5%	$\geq$ 350 F
2. STARTUP	$\geq$ 0.99	$\leq$ 5%	$\geq$ 350 F
3. HOT STANDBY	< 0.99	0	$\geq$ 350 F
4. HOT SHUTDOWN	< 0.99	0	350 F > T_{avg} > 200 F
5. COLD SHUTDOWN	< 0.99	0	$\leq$ 200 F
6. REFUELING**	$\leq$ 0.90	0	$\leq$ 140 F

* Excluding decay heat.

** Fuel in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

SECTION 2.0

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop average coolant temperature (T_{avg}) shall not exceed the limits shown in Figure 2.1-1, for three loop operation.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour, and comply with the requirements of Specification 6.7.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

MODES 1 and 2:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour, and comply with the requirements of Specification 6.7.1.

MODES 3, 4 and 5:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes, and comply with the requirements of Specification 6.7.1.

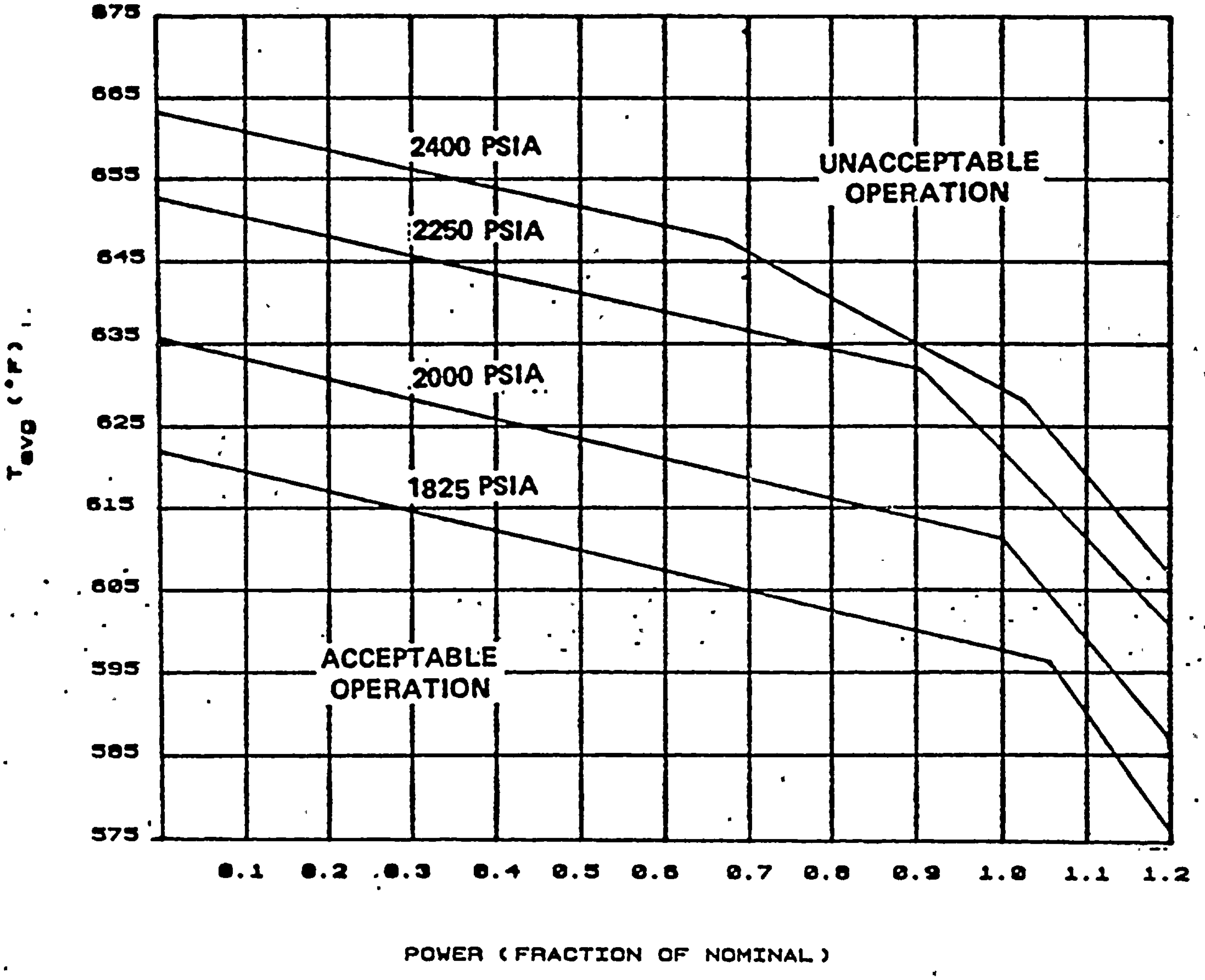

FIGURE 2.1-1

REACTOR CORE SAFETY LIMIT

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETINGS

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The Reactor Trip System Instrumentation and Interlock Setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

a. With a Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Trip Setpoint column but more conservative than the value shown in the Allowable Value column of Table 2.2-1, adjust the Setpoint consistent with the Trip Setpoint value.

b. With the Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Allowable Value column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its Setpoint adjusted consistent with the Trip Setpoint value.

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Power Range, Neutron Flux High Setpoint	$\leq$ 109% of RTP**	$\leq$ 110% of RTP**
Low Setpoint	$\leq$ 25% of RTP**	$\leq$ 26% of RTP**
3. Intermediate Range, Neutron Flux	$\leq$ 25% of RTP**	$\leq$ 30% of RTP**
4. Source Range, Neutron Flux	$\leq 10^5$ cps	$\leq 1.3 \times 10^5$ cps
5. Overtemperature ΔT	See Note 1	See Note 2
6. Overpower ΔT	See Note 3	See Note 4
7. Pressurizer Pressure-Low	$\geq$ 1835 psig	$\geq$ 1825 psig
8. Pressurizer Pressure-High	$\leq$ 2385 psig	$\leq$ 2395 psig
9. Pressurizer Water Level-High	$\leq$ 92% of inst. span	$\leq$ 93% of inst. span
10. Reactor Coolant Flow-Low	$\geq$ 90% of loop design flow*	$\geq$ 89% of loop design flow*
11. Steam Generator Water Level Low-Low	$\geq$ 15% of narrow range inst. span	$\geq$ 14% of narrow range inst. span
12. Steam/Feedwater Flow Mismatch Coincident With	Feed flow $\leq 0.64 \times 10^6$ lb/hr below steam flow	Feed Flow $\leq 0.72 \times 10^6$ lb/hr below steam flow
Steam Generator Water Level-Low	$\geq$ 15% of narrow range inst. span	$\geq$ 14% of narrow range inst. span

*Loop design flow = 89,500 gpm
**RTP - RATED THERMAL POWER

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
13. Undervoltage - 4Kv Bus	$\geq$ 2496 volts - each bus	$\geq$ 2456 volts- each bus
14. Underfrequency - Reactor Coolant Pump Breakers	$\geq$ 56.1 Hz	$\geq$ 56.0 hz.
15. Turbine Trip		
Auto Stop Oil Pressure	$\geq$ 45 psig	$\geq$ 40 psig
Turbine Stop Valve Closure	Not Applicable	Not Applicable
16. Safety Injection Input from ESF	Not Applicable	Not Applicable
17. Reactor Trip System Interlocks		
a. Intermediate Range Neutron Flux, P-6	$\geq 1 \times 10^{-10}$ amp	$\geq 6 \times 10^{-11}$ amp.
b. Low Power Reactor Trips Block, P-7	10% of RTP** and $\leq$ 10% turbine power	$\geq$ 9%, < 11% of RTP** $\leq$ 11% turbine power
c. Power Range Neutron Flux, P-8	$\leq$45% of RTP**	$\leq$46% of RTP **
d. Power Range Neutron Flux, P-10	10% of RTP**	$\geq$9%, < 11% of RTP **
18. Reactor Coolant Pump Breaker Position	Not Applicable	Not Applicable
19. Reactor Trip Breakers	Not Applicable	Not Applicable
20. Automatic Trip and Interlock Logic	Not Applicable	Not Applicable

**RTP - RATED THERMAL POWER

TABLE 2.2-1 (Continued)

TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE Δ T

$$\Delta T \left[\frac{1}{1+\tau_1 S}\right] \leq \Delta T_o \left\{K_1 - K_2 \frac{(1+\tau_2 S)}{(1+\tau_3 S)} [T \left[\frac{1}{1+\tau_4 S}\right] - T'] + K_3 (P-P') - f(\Delta I)\right\}$$

Where:

ΔT	=	Measured ΔT by RTD Bypass Loop Instrumentation:
$\frac{1}{1+\tau_1 S}$	=	Lag compensator on measured ΔT ;
τ_1	=	Time constants utilized in the lag compensator for $\Delta T, \tau_1 = 2$ s;
ΔT_o	=	Indicated ΔT at RATED THERMAL POWER;
K_1	=	1.095;
K_2	=	0.0107/ F;
$\frac{1+\tau_2 S}{1+\tau_3 S}$	=	The function generated by the lead-lag compensator for T_{avg} dynamic compensation;
τ_2, τ_3	=	Time constants utilized in the Lead-lag compensator for T_{avg}, $\tau_2 = 25$ s, $\tau_3 = 3$ s;
T	=	Average temperature, F;
$\frac{1}{1+\tau_4 S}$	=	Lag compensator on measured T_{avg} lag compensator;
τ_4	=	Time constant utilized in the measured T_{avg} lag compensator, $\tau_4 = 2$ s;

TABLE 2.2-1 (Continued)

TABLE NOTATIONS Continued

NOTE 1: (Continued)

T'	$\leq$	574.2 F (Nominal T_{avg} at RATED THERMAL POWER);
K_3	=	0.000453/psig;
p	=	Pressurizer pressure, psig;
p'	=	2235 psig (Nominal RCS operating pressure);
S	=	Laplace transform operator, 1/second

and f (ΔI) is a function of the indicated difference between top and bottom detectors of the power-range neutron ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(1) for $q_t - q_b$ between - 14% and + 10%, f (ΔI) = 0, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(2) For each percent that the magnitude of $q_t - q_b$ exceeds -14%, the ΔT Trip Setpoint shall be automatically reduced by 2.0% of its value at RATED THERMAL POWER; and

(3) For each percent that the magnitude of $q_t - q_b$ exceeds +10%, the ΔT Trip Setpoint shall be automatically reduced by 3.5% of its value at RATED THERMAL POWER.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 2%.

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (CONTINUED)

NOTE 3: OVERPOWER ΔT

$$\Delta T \left(\frac{1}{1+\tau_1 S}\right) \leq \Delta T_o \{K_4\ K_5 \left[\frac{\tau_5 S}{1+\tau_5 S}\right]\left[\frac{1}{1+\tau_4 S}\right] T - K_6 \left[T \left(\frac{1}{(1+\tau_4 S)}\right) - T''\right] - f(\Delta I)\}$$

Where:	ΔT	=	As defined in Note 1,
	$\frac{1}{1+\tau_1 S}$	=	As defined in Note 1,
	τ_1	=	As defined in Note 1,
	ΔT_o	=	As defined in Note 1,
	K_4	=	1.09;
	K_5	=	0.02/ °F; for increasing average temperature and 0 for decreasing average temperature,
	$\frac{\tau_5 S}{1+\tau_5 S}$	=	The function generated by the rate-lag compensator for T_{avg} dynamic compensation;
	τ_5	=	Time constants utilized in the rate-lag compensator for T_{avg}, τ_5 = 10 s,
	$\frac{1}{1+\tau_4 S}$	=	As defined in Note 1,
	τ_4	=	As defined in Note 1,

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: (Continued)

K_6	=	0.00068/ F for T > T" and K_6 = 0 for T ≤ T ",
T	=	As defined in Note 1,
T"	=	Indicated T_{avg} at RATED THERMAL POWER
S	=	As defined in Note 1, and
f (ΔI)	=	As defined in Note 1

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 2%.

BASES

FOR

SECTION 2.0

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

NOTE

The BASES contained in succeeding pages summarize the reasons for the Specifications in Section 2.0, but in accordance with 10 CFR 50.36 are not part of these Technical Specifications.

2.1.1 REACTOR CORE

The restrictions of this Safety Limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and reactor coolant temperature and pressure have been related to DNB. This relationship has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location, to the local heat flux, is indicative of the margin to DNB.

The DNB design basis is as follows: there must be at least a 95 percent probability with 95 percent confidence that the minimum DNBR of the limiting rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used. The correlation DNBR limit is established based on the entire applicable experimental data set such that there is a 95 percent probability with 95 percent confidence that DNB will not occur when the minimum DNBR is at the DNBR limit.

The curves of Figure 2.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than the design DNBR value, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid.

These curves are based on an enthalpy hot channel factor, $F^N_{\Delta H}$, of 1.62 and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:

$$F^N_{\Delta H} \leq 1.62 \, [1 + 0.3 \, (1-P)]$$

Where P is the fraction of RATED THERMAL POWER.

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion limit assuming the axial power imbalance is within the limits of

2.1.1 REACTOR CORE (Continued)

the f(Δq) function of the Overtemperature ΔT trip. When the axial power imbalance is not within tolerance, the axial power imbalance effect on the Overtemperature ΔT trips will reduce the setpoints to provide protection consistent with core safety limits.

Fuel rod bowing reduces the values of DNB ratio (DNBR). The amount of the DNBR reduction is 4.7% for LOPAR fuel with the L-grid DNB correlation and 5.5% for the OFA fuel with the WRB-1 DNB correlation. The penalties are calculated persuant to "Fuel Rod Bow Evaluation," WCAP-8691-P-A Revision 1 (Proprietary) and WCAP 8692 Revision 1 (non-proprietary). The restrictions of the Core Thermal Hydraulic Safety Limits assure that an amount of DNBR margin greater than or equal to the above penalties is retained to offset the rod bow DNBR penalty.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System (RCS) from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor vessel, pressurizer, and the RCS piping, valves and fitting are designed to ANSI B31.1 which permits a maximum transient pressure of 120% of design pressure. Section III of the ASME Code for Nuclear Power Plants permits a maximum transient pressure of 110% (2735 psig) of the design pressure of 2485 psig. The Safety Limit of 2735 psig is therefore consistent with the design criteria and associated Code requirements.

[The entire RCS was hydrotested at 125% (3110 psig) of design pressure, to demonstrate integrity prior to initial operation.]

2.2 LIMITING SAFETY SYSTEM SETTINGS

BASES

2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Trip Setpoint Limits specified in Table 2.2-1 are the nominal values at which the Reactor Trips are set for each functional unit. The Trip Setpoints have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their safety limits during normal operation and design basis anticipated operational occurrences and to assist the Engineered Safety Features Actuation System in mitigating the consequences of accidents. The various reactor trip circuits automatically open the reactor trip breakers whenever a condition monitored by the Reactor Protection System reaches a preset or calculated level. In addition to redundant channels and trains, the design approach provides a Reactor Protection System which monitors numerous system variables, therefore, providing protection system functional diversity.

The Reactor Protection System initiates a turbine trip signal whenever reactor trip is initiated. This prevents the reactivity insertion that would otherwise result from excessive reactor system cooldown and thus avoids unnecessary actuation of the Engineered Safety Features Actuation System.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance for all trips including those trips assumed in the safety analyses.

Manual Reactor Trip

The Reactor Trip System includes manual Reactor trip capability.

Power Range, Neutron Flux

In each of the Power Range Neutron Flux channels there are two independent bistables, each with its own trip setting used for a High and Low Range trip setting. The Low Setpoint trip provides protection during subcritical and low power operations to mitigate the consequences of a power excursion beginning from low power, and the High Setpoint trip provides protection during power operations for all power levels to mitigate the consequences of a reactivity excursion which may be too rapid for the temperature and pressure protective trips.

The Low Setpoint trip may be manually blocked above P-10 (a power level of approximately 10% of RATED THERMAL POWER) and is automatically reinstated below the P-10 Setpoint.

Intermediate and Source Range, Neutron Flux

The Intermediate and Source Range, Neutron Flux trips provide core protection during reactor startup to mitigate the consequences of an uncontrolled rod cluster control assembly bank withdrawal from a subcritical condition. These trips provide redundant protection to the Low Setpoint trip of the Power Range, Neutron Flux channels. The Source Range channels will initiate a Reactor trip at about 10^5 counts per second unless manually blocked when P-6 becomes active. The Intermediate Range channels will initiate a Reactor trip at a current level equivalent to approximately 25% of RATED THERMAL POWER unless manually blocked when P-10 becomes active.

Overtemperature ΔT

The Overtemperature ΔT trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 4 seconds), and pressure is within the range between the Pressurizer High and Low Pressure trips. The Setpoint is automatically varied with: (1) coolant temperature to correct for temperature induced changes in density and heat capacity of water and includes dynamic compensation for piping delays from the core to the loop temperature detectors, (2) pressurizer pressure, and (3) axial power distribution. With normal axial power distribution, this Reactor trip limit is always below the core Safety Limit as shown in Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the Reactor trip is automatically reduced according to the notations in Table 2.2-1.

Overpower ΔT

The Overpower ΔT trip prevents power density anywhere in the core from exceeding 118% of design power density. This provides assurance of fuel integrity (e.g., no fuel pellet melting and less than 1% cladding strain under all possible overpower conditions, limits the required range for Overtemperature ΔT trip, and provides a backup to the High Neutron Flux trip. The Setpoint is automatically varied with: (1) coolant temperature to correct for temperature induced changes in density and heat capacity of water, (2) rate of change of temperature for dynamic compensation for piping delays from the core to the loop temperature detectors, and (3) axial power distribution to ensure that the allowable heat generation rate (kW/ft) is not exceeded.

Pressurizer Pressure

In each of the pressurizer pressure channels, there are two independent bistables, each with its own trip setting to provide for a High and Low Pressure trip thus limiting the pressure range in which reactor operation is permitted. The Low Setpoint trip protects against low pressure which could lead to DNB by tripping the reactor in the event of a loss of reactor coolant pressure.

On decreasing power the Low Setpoint trip is automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER with turbine impulse chamber pressure at approximately 10% of full power equivalent); and on increasing power, automatically reinstated by P-7.

The High Setpoint trip functions in conjunction with the pressurizer relief and safety valves to protect the Reactor Coolant System against system overpressure.

Pressurizer Water Level

The Pressurizer High Water Level trip is provided to prevent water relief through the pressurizer safety valves. On decreasing power the Pressurizer High Water Level trip is automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER with a turbine impulse chamber pressure at approximately 10% of full power equivalent); and on increasing power, automatically reinstated by P-7.

Reactor Coolant Flow

The Low Reactor Coolant Flow trips provide core protection to prevent DNB by mitigating the consequences of a loss of flow resulting from the loss of one or more reactor coolant pumps.

On increasing power above P-7 (a power level of approximately 10% of RATED THERMAL POWER or a turbine impulse chamber pressure at approximately 10% of full power equivalent), an automatic Reactor trip will occur if the flow in more than one loop drops below 90% of nominal full loop flow. Above P-8 (a power level of approximately 45% of RATED THERMAL POWER) an automatic Reactor trip will occur if the flow in any single loop drops below 90% of nominal full loop flow. Conversely, on decreasing power between P-8 and P-7 an automatic Reactor trip will occur on low reactor coolant flow in more than one loop and below P-7 the trip function is automatically blocked.

Reactor Coolant Flow Cont.

Under-frequency sensors are also installed on the 4 Kv buses to detect under-frequency and initiate a RCP breaker trip on under-frequency. The under-frequency trip setpoint preserves the coastdown energy of the reactor coolant pumps, in case of a grid frequency decrease, so DNB does not occur.

Steam Generator Water Level

The Steam Generator Water Level Low-Low trip protects the reactor from loss of heat sink in the event of a sustained steam/feedwater flow mismatch resulting from loss of normal feedwater. The specified Setpoint provides allowances for starting delays of the Auxiliary Feedwater System.

Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level

The Steam/Feedwater Flow Mismatch in coincidence with a Steam Generator Low Water Level trip is not used in the transient and accident analyses but is included in Table 2.2-1 to ensure the functional capability of the specified trip settings and thereby enhance the overall reliability of the Reactor Trip System. This trip is redundant to the Steam Generator Water Level Low-Low trip. The Steam/Feedwater Flow Mismatch portion of this trip is activated when the steam flow exceeds the feedwater flow by greater than or equal to 0.64×10^6 lbs/hour. The Steam Generator Low Water level portion of the trip is activated when the water level drops below 15%, as indicated by the narrow range instrument. These trip values include sufficient allowance in excess of normal operating values to preclude spurious trips but will initiate a Reactor trip before the steam generators are dry. Therefore, the required capacity and starting time requirements of the auxiliary feedwater pumps are reduced and the resulting thermal transient on the Reactor Coolant System and steam generators is minimized.

Undervoltage - 4Kv Bus Trips

The 4Kv Bus Undervoltage trips provide core protection against DNB as a result of complete loss of forced coolant flow. The specified Setpoint assures a Reactor trip signal is generated before the Low Flow Trip Setpoint is reached. Time delays are incorporated in the Undervoltage trips to prevent spurious Reactor trips from momentary electrical power transients. The delay is set so that the time required for a signal to reach the Reactor trip breakers following the simultaneous trip of two or more reactor coolant pump bus circuit breakers shall not exceed 1.3

Undervoltage - 4Kv Bus Trips Cont.

seconds. On decreasing power the Undervoltage Bus trips are automatically blocked by P-7 (a power level of approximataly 10% of RATED THERMAL POWER with a turbine impulse chamber pressure at approximately 10% of full power equivalent); and on increasing power, reinstated automatically by P-7.

Turbine Trip

A Turbine Trip initiates a reactor trip. On decreasing power the turbine trip is automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER with a turbine impulse chamber at approximately 10% of full power equivalent); and on increasing power, reinstated automatically by P-7.

Safety Injection Input from ESF

If a Reactor trip has not already been generated by the Reactor Trip System instrumentation, the ESF automatic actuation logic channels will initiate a Reactor trip upon any signal which initiates a Safety Injection. The ESF instrumentation channels which initiate a Safety Injection signal are shown in Table 3.3-3.

Reactor Coolant Pump Breaker Position Trip

The Reactor Coolant Pump Breaker Position Trips are anticipatory trips which provide reactor core protection against DNB. The open/close position trips assure a reactor trip signal is generated before the low flow trip setpoint is reached. No credit was taken in the accident analyses for operation of these trips. Their functional capability at the open/close position settings is required to enhance the overall realiability of the Reactor Protection System. Above P-7 (a power level of approximataly 10% of RATED THERMAL POWER or a turbine impulse chamber pressure at approximataly 10% of full power equivalent) an automatic reactor trip will occur if more than one reactor coolant pump breaker is opened. Above P-8 (a power level of approximataly 45% of RATED THERMAL POWER) an automatic reactor trip will occur if one reactor coolant pump breaker is opened. On decreasing power between P-8 and P-7 an automatic reactor trip will occur if more than one reactor coolant pump breaker is opened and below P-7 the trip function is automatically blocked.

Reactor Trip System Interlocks

The Reactor Trip System interlocks perform the following functions:

P-6 On increasing power P-6 allows the manual block of the Source Range trip (i.e., prevents premature block of Source Range trip) and deenergizes the high voltage to the detectors. On decreasing power, Source Range Level trips are automatically reactivated and high voltage restored.

P-7 On increasing power P-7 automatically enables reactor trips on low flow in more than one primary coolant loop, more than one reactor coolant pump breaker open, reactor coolant pump bus under-voltage, turbine trip, pressurizer low pressure and pressurizer high level. On decreasing power the above listed trips are automatically blocked.

P-8 On increasing power, P-8 automatically enables Reactor trip on low flow in one or more reactor coolant loops, and one or more reactor coolant pump breakers open. On decreasing power, the P-8 interlock automatically blocks the trip on low flow in one coolant loop or one coolant pump breaker open.

P-10 On increasing power, P-10 allows the manual block of the Intermediate Range trip and the Low Setpoint Power Range trip; and automatically deenergizes the Source Range high voltage power. On decreasing power, the Intermediate Range trip and the Low Setpoint Power Range trip are automatically reactivated. P-10 also provides an input to P-7.

SECTIONS 3.0 AND 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.0 APPLICABILITY

LIMITING CONDITION FOR OPERATION

3.0.1 Compliance with the Limiting Conditions for Operation contained in the succeeding specifications is required during the OPERATIONAL MODES or other conditions specified therein; except that upon failure to meet the Limiting Conditions for Operation, the associated ACTION requirements shall be met.

3.0.2 Noncompliance with a specification shall exist when the requirements of the Limiting Condition for Operation and associated ACTION requirements are not met within the specified time intervals. If the Limiting Condition for Operation is restored prior to expiration of the specified time intervals, completion of the ACTION requirements is not required.

3.0.3 When a Limiting Condition for Operation is not met, except as provided in the associated ACTION requirements, within 1 hour action shall be initiated to place the unit in a MODE in which the specification does not apply by placing it, as applicable, in:

a. At least HOT STANDBY within the next 6 hours,

b. At least HOT SHUTDOWN within the following 6 hours, and

c. At least COLD SHUTDOWN within the subsequent 24 hours.

Where corrective measures are completed that permit operation under the ACTION requirements, the action may be taken in accordance with the specified time limits as measured from the time of failure to meet the Limiting Condition for Operation. Exceptions to these requirements are stated in the individual specifications.

This specification is not applicable in Mode 5 or 6.

3.0.4 Entry into an OPERATIONAL MODE or other specified condition shall not be made unless the conditions for the Limiting Condition for Operation are met without reliance on provisions contained in the ACTION requirements. This provision shall not prevent passage through or to OPERATIONAL MODES as required to comply with ACTION requirements. Exceptions to these requirements are stated in the individual specifications.

3.0.5 For purposes of determining if a component is operable for Limiting Conditions for Operation considerations, the component need not be considered inoperable due to inoperability of its normal or emergency power supply if all of its redundant components are operable with their normal or emergency power supplies operable.

SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Requirements shall be met during the OPERATIONAL MODES or other conditions specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement.

4.0.2 Each Surveillance Requirement shall be performed within the specified time interval with:

a. A maximum allowable extension not to exceed 25% of the surveillance interval, but

b. The combined time interval for any three consecutive surveillance intervals shall not exceed 3.25 times the specified surveillance interval.

4.0.3 Failure to perform a Surveillance Requirement within the specified time interval shall constitute a failure to meet the OPERABILITY requirements for a Limiting Condition for Operation. Exceptions to these requirements are stated in the individual specifications. Surveillance Requirements do not have to be performed on inoperable equipment.

4.0.4 Entry into an OPERATIONAL MODE or other specified condition shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation has been performed within the stated surveillance interval or as otherwise specified.

4.0.5 Surveillance Requirements for inservice inspection and testing of ASME Code Class 1, 2, and 3 components shall be applicable as follows:

a. Inservice inspection of ASME Code Class 1, 2, and 3 components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50, Section 50.55a(g), as provided in the Turkey Point Inservice Inspection and Inservice Testing Programs.

4.0.5 (Continued)

b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing activities	Required frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days

c. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice inspection and testing activities;

d. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements; and

e. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

SECTIONS 3.0 AND 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{avg} GREATER THAN 200 F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 1.77% $\Delta k/k$.

APPLICABILITY: MODES 1, 2*, 3, and 4.

ACTION:

With the SHUTDOWN MARGIN less than 1.77% $\Delta k/k$, immediately initiate and continue boration at greater than or equal to 10 gpm of a solution containing greater than or equal to 20,000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1.77% $\Delta k/k$:

a. Within 1 hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s);

b. When in MODE 1 or MODE 2 with k_{eff} greater than or equal to 1 at least once per 12 hours by verifying that control bank withdrawal is within the limits of Specification 3.1.3.6.;

c. When in MODE 2 with k_{eff} less than 1, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.6;

*See Special Test Exceptions Specification 3.10.1.

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of Specification 4.1.1.1.1e. below, with the control banks at the maximum insertion limit of Specification 3.1.3.6; and

e. When in MODE 3 or 4, at least once per 24 hours by consideration of the following factors:

1) Reactor Coolant System boron concentration,

2) Control rod position,

3) Reactor Coolant System average temperature,

4) Fuel burnup based on gross thermal energy generation,

5) Xenon concentration, and

6) Samarium concentration.

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1% $\Delta k/k$ at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1e, above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 EFPD after each fuel loading. If the difference between the observed and predicted steady state values reaches the equivalent of 1% $\Delta k/k$, make an evaluation as to the cause of the discrepancy and prepare and submit a Report to the Commission pursuant to Specification 6.9.2 within 30 days.

SHUTDOWN MARGIN - T_{avg} LESS THAN OR EQUAL TO 200 F

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to 1% Δk/k.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than 1% Δk/k, immediately initiate and continue boration at greater than or equal to 10 gpm of a solution containing greater than or equal to 20,000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1% Δk/k:

a. Within 1 hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s); and

b. At least once per 24 hours by consideration of the following factors:

   1) Reactor Coolant System boron concentration,

   2) Control rod position,

   3) Reactor Coolant System average temperature,

   4) Fuel burnup based on gross thermal energy generation,

   5) Xenon concentration, and

   6) Samarium concentration.

REACTIVITY CONTROL SYSTEMS

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be:

a. Equal to or less positive than 5.0×10^{-5} $\Delta k/k/$ F for the all rods withdrawn, beginning of cycle life (BOL), less than or equal to 70% of RATED THERMAL POWER condition; and

b. Equal to or less positive than 5.0×10^{-5} $\Delta k/k/$ F for the all rods withdrawn, beginning of cycle life (BOL)decreasing linearly to zero from 70% to 100% of RATED THERMAL POWER condition and

c. Less negative than -3.5×10^{-4} $\Delta k/k/$ F for the all rods withdrawn, end of cycle life (EOL), RATED THERMAL POWER condition.

APPLICABILITY: Specification 3.1.1.3a. and b. - MODES 1 and 2* only.**
Specification 3.1.1.3c. - MODES 1, 2, and 3 only.**

ACTION:

a. With the MTC more positive than the limit of Specification 3.1.1.3a. or b. above, operation in MODES 1 and 2 may proceed provided:

1. Control rod withdrawal limits are established and maintained sufficient to restore the MTC to its limit within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6;

2. The control rods are maintained within the withdrawal limits established above until a subsequent calculation verifies that the MTC has been restored to within its limit for the all rods withdrawn condition; and

*With K_{eff} greater than or equal to 1.
**See Special Test Exceptions Specification 3.10.3.

LIMITING CONDITION FOR OPERATION

ACTION (Continued):

3. A Special Report is prepared and submitted to the Commission, pursuant to Specification 6.9.2, within 10 days, describing the value of the measured MTC, the interim control rod withdrawal limits, and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.

b. With the MTC more negative than the limit of Specification 3.1.1.3c. above, be in HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.1.1.3 The MTC shall be determined to be within its limits during each fuel cycle as follows:

a. The MTC shall be measured and compared to the BOL limit of Specification 3.1.1.3a. and b., above, prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading; and

b. The MTC shall be measured at any THERMAL POWER and compared to -3.0×10^{-4} $\Delta k/k/$ F (all rods withdrawn, RATED THERMAL POWER condition) within 7 EFPD after reaching an equilibrium boron concentration of 300 ppm. In the event this comparison indicates the MTC is more negative than -3.0×10^{-4} $\Delta k/k/$ F, the MTC shall be remeasured, and compared to the EOL MTC limit of Specification 3.1.1.3c., at least once per 14 EFPD during the remainder of the fuel cycle.

MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

3.1.1.4 The Reactor Coolant System lowest operating loop temperature (T_{avg}) shall be greater than or equal to 541 F.

APPLICABILITY: MODES 1 and 2* **.

ACTION:

With a Reactor Coolant System operating loop temperature (T_{avg}) less than 541 F, restore T_{avg} to within its limit within 15 minutes or be in HOT STANDBY within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.1.1.4 The Reactor Coolant System temperature (T_{avg}) shall be determined to be greater than or equal to 541 F:

a. Within 15 minutes prior to achieving reactor criticality, and

b. At least once per 30 minutes when the reactor is critical and the Reactor Coolant System T_{avg} is less than 547 F with the T_{avg} -T_{ref} Deviation Alarm not reset.

*With k_{eff} greater than or equal to 1

**See Special Test Exceptions Specification 3.10.3.

3/4.1.2 BORATION SYSTEMS

FLOW PATH - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.1 As a minimum, one of the following boron injection flow paths shall be OPERABLE and capable of being powered from an OPERABLE emergency power source.

a. A flow path from the boric acid tanks via a boric acid transfer pump and a charging pump to the Reactor Coolant System if the boric acid storage tank in Specification 3.1.2.5a is OPERABLE, or

b. The flow path from the refueling water storage tank via a charging pump to the Reactor Coolant System if the refueling water storage tank in Specification 3.1.2.5b is OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With none of the above flow paths OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.1 At least one of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that the temperature of the heat traced portion of the flow.path is greater than or equal to 145 F when a flow path from the boric acid tanks is used, and

b. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position, and that each pump in the flow path is capable of being powered from an OPERABLE emergency power source.

REACTIVITY CONTROL SYSTEMS

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 At least two of the following three boron injection flow paths shall be OPERABLE:

a. One flow path from the boric acid tanks via a boric acid transfer pump and a charging pump to the Reactor Coolant System (RCS), and

b. Two flow paths from the refueling water storage tank via charging pumps to the RCS.

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

With only one of the above required boron injection flow paths to the RCS OPERABLE, restore at least two boron injection flow paths to the RCS to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least $1\% \cdot \Delta k/k$ at 200 F within the next 6 hours; restore at least two flow paths to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.2 At least two of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that the temperature of the heat traced portion of the flow path from the boric acid tanks is greater than or equal to 145 F when it is a required water source:

b. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position;

c. At least once per refueling by verifying that the flow path required by Specification 3.1.2.2a delivers at least 10 gpm to the RCS.

CHARGING PUMP - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.3 One charging pump in the boron injection flow path required by Specification 3.1.2.1 shall be OPERABLE (and capable of being powered from an OPERABLE emergency power source).

APPLICABILITY: MODES 5 and 6.

ACTION:

With no charging pump OPERABLE (or capable of being powered from an OPERABLE emergency power source) suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.3 At least once per 31 days one charging pump shall be started and its flow verified to be at least 45 gpm.

REACTIVITY CONTROL SYSTEMS

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two charging pumps with independent power supplies shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one charging pump OPERABLE, restore at least two charging pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1% $\Delta k/k$ at 200 F within the next 6 hours; restore at least two charging pumps to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.4 At least two charging pumps shall be demonstrated OPERABLE by testing pursuant to Specification 4.0.5. The provisions of Specification 4.0.4 are not applicable for entry Modes 3 and 4.

BORATED WATER SOURCE - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.5 As a minimum, one of the following borated water sources shall be OPERABLE:

a. A Boric Acid Storage System with:

1) A minimum contained borated water volume of 1,154 gallons/unit,

2) A boron concentration of 20,000 ppm to 22,500 ppm

3) A minimum solution temperature of 145 F

b. The refueling water storage tank (RWST) with:

1) A minimum contained borated water volume of 19,455 gallons/unit,

2) A minimum boron concentration of 1950 ppm, and

3) A minimum solution temperature of 39 F

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water source OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.5 The above required borated water source shall be OPERABLE:

a. At least once per 7 days by:

1) Verifying the boron concentration of the water,

2) Verifying the contained borated water volume, and

3) Verifying the boric acid storage tank solution temperature when it is the source of borated water

b. At least once per 24 hours by verifying the RWST temperature when it is the source of borated water and the outside air temperature is less than 39 F.

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.6 As a minimum, the following borated water source(s) shall be OPERABLE as required by Specification 3.1.2.2:

a. A Boric Acid Storage System with:

1) A minimum contained borated water volume of 3080 gallons per unit,

2) A boron concentration of 20,000 ppm to 22,500 ppm

3) A minimum solution temperature of 145 F

b. The refueling water storage tank (RWST) with:

1) A minimum contained borated water volume of 320,000 gallons,

2) A minimum boron concentration of 1950 ppm, and

3) A minimum solution temperature of 39 F

4) A maximum solution temperature of 100 F

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

a. With the Boric Acid Storage System inoperable and being used as one of the above required borated water sources, restore the system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 1% Δ k/k at 200 F; restore the Boric Acid Storage System to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the following 30 hours.

b. With the RWST inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.6 Each borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the boron concentration of the water,

2) Verifying the contained borated water volume of the water source, and

3) Verifying the boric acid storage tank solution temperature when it is the source of borated water.

b. At least once per 24 hours by verifying the RWST temperature when the outside air temperature is either less than 39 F or greater than 100 F.

HEAT TRACING

LIMITING CONDITION FOR OPERATION

3.1.2.7 At least two channels of heat tracing shall be OPERABLE for the heat traced portions of the boric acid flow paths required by Specification 3.1.2.2.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With only one channel of heat tracing OPERABLE, operation may continue for up to 30 days provided the flow path temperatures are verified to be greater than or equal to 145 F at least once per 8 hours; otherwise, be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.7 Each heat tracing channel for the boric acid flow path shall be demonstrated OPERABLE:.

a. At least once per 31 days by energizing each heat tracing channel, and

b. At least once per 24 hours by verifying the flow path temperatures to be greater than or equal to 145 F. The flow path temperature shall be determined by either measurement or recirculation flow until establishment of equilibrium temperatures within the tank.

REACTIVITY CONTROL SYSTEMS

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

3.1.3.1 All full length (shutdown and control) rods, shall be OPERABLE and positioned within ± 12 steps (indicated position) of the REFERENCE POSITION corresponding to the group demand counter position within one hour after rod motion.

APPLICABILITY: MODES 1* and 2*.

ACTION:

a. With one or more full length rods inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour and be in HOT STANDBY within 6 hours.

b. With one full rod inoperable due to causes other than addressed by ACTION a, above, or misaligned from its REFERENCE POSITION by more than ± 12 steps (indicated position), POWER OPERATION may continue provided that within 1 hour either:

   1. The rod is restored to OPERABLE status within the above alignment requirements, or

   2. The remainder of the rods in the bank with the inoperable rod are aligned to within ± 12 steps of the inoperable rod while maintaining the rod sequence and insertion limits of Figure 3.1-1; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation, or

   3. The rod is declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. POWER OPERATION may then continue provided that:

*See Special Test Exceptions 3.10.2 and 3.10.3.

ACTION (Continued):

a. A re-evaluation of each accident analysis of Table 3.1-1 is performed within 5 days; this re-evaluation shall confirm that the previously analyzed results of these accidents remain valid for the duration of operation under these conditions.

b. The SHUTDOWN MARGIN requirements of Specification 3.1.1.1 is determined at least once per 12 hours.

c. A power distribution map is obtained from the movable incore detectors and $F_Q(Z)$ and $F^N_{\Delta H}$ are verified to be within their limits within 72 hours.

d. The THERMAL POWER level is reduced to less than or equal to 75% of RATED THERMAL POWER within one hour and within the next 4 hours the high neutron flux trip setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER. THERMAL POWER shall be maintained less than or equal to 75% of RATED THERMAL POWER until compliance with ACTIONS 3.1.3.1.b.3.a and 3.1.3.1.b.3.c above are demonstrated.

c. With more than one rod trippable but inoperable due to causes other than addressed by ACTION a. above, POWER OPERATION may continue provided that:

1. Within one hour, the remainder of the rods in the bank(s) with the inoperable rods are aligned to within ± 12 steps of the inoperable rods while maintaining the rod sequence and insertion limits of Figure 3.1-1. The THERMAL POWER Level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation, and

2. The inoperable rods are restored to OPERABLE status within 72 hours.

d. With more than one rod misaligned from its reference position by more than ± 12 steps (indicated position), be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each full rod shall be determined to be within ± 12 steps (indicated position) of the REFERENCE POSITION corresponding to the group demand position at least once per 12 hours (allowing for one hour thermal soak after rod motion) except during time intervals when the Rod Position Deviation Monitor is inoperable, then verify the group positions at least once per 4 hours.

4.1.3.1.2 Each full length rod not fully inserted which is inserted in the core shall be determined to be OPERABLE by movement of at least 10 steps in any one direction at least once per 31 days.

TABLE 3.1-1

ACCIDENT ANALYSES REQUIRING RE-EVALUATION IN THE EVENT OF AN INOPERABLE FULL-LENGTH ROD

Rod Cluster Control Assembly Insertion Characteristics

Rod Cluster Control Assembly Misalignment

Loss of Reactor Coolant from Small Ruptured Pipes or from Cracks in Large Pipes Which Actuates the Emergency Core Cooling System

Single Rod Cluster Control Assembly Withdrawal at Full Power

Major Reactor Coolant System Pipe Ruptures (Loss of Coolant Accident)

Major Secondary Coolant System Pipe Rupture

Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)

REACTIVITY CONTROL SYSTEMS

POSITION INDICATION SYSTEMS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.3.2 The shutdown control individual rod position indication system and the demand position indication system shall be OPERABLE and capable of determining the actual and demanded control rod positions, respectively, as follows:

Analog rod position indicators, within one hour after rod motion (allowance for thermal soak):

<u>All Shutdown Banks</u> - ± 12 steps of the group demand counters for withdrawal ranges of 0-30 steps and 200-228 steps.

<u>Control Banks A and B</u> - ± 12 steps of the group demand counters for withdrawal ranges of 0-30 steps and 200-228 steps.

<u>Control Banks C and D</u> - ± 12 steps of the group demand counters for withdrawal ranges of 0-30 steps and 150-228 steps. ± 12 steps of the REFERENCE POSITION for withdrawal range of 31-149 steps.

Group demand counters: ± 2 steps

<u>APPLICABILITY</u>: MODES 1 and 2

<u>ACTION</u>:

a. With a maximum of one analog rod position indicator per bank inoperable either:

   1. Determine the position of the non-indicating rod(s) indirectly by the movable incore detectors at least once per 8 hours and within one hour after any motion of the non-indicating rod which exceeds 24 steps in one direction since the last determination of the rod's position, or

   2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 8 hours.

b. With a maximum of one group demand position indicator per bank inoperable either:

   1. Verify that all rod position indicators for the affected bank are OPERABLE and that the most withdrawn rod and the least withdrawn rod of the bank are within a maximum of 12 steps (corrected indicated position) of each other at least once per 8 hours, or

LIMITING CONDITION FOR OPERATION (continued)

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 8 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.2.1 Each analog rod position indicator shall be determined to be OPERABLE by verifying that the demand position indication system and the rod position indication system (by use of the REFERENCE POSITION) agree within 12 steps (allowing for one hour thermal soak after rod motion) at least once per 12 hours except during time intervals when the Rod Position Deviation Monitor is inoperable, then compare the demand position indication system and the rod position indication system (by use of the REFERENCE POSITION) at least once per 4 hours.

4.1.3.2.2 Each of the above required rod position indicator(s) shall be determined to be OPERABLE by performance of a CHANNEL FUNCTIONAL TEST at least once per 18 months.

POSITION INDICATION SYSTEM - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.3.3 The group demand position indicator shall be OPERABLE and capable of determining within ± 2 steps the demand position of each shutdown or control rod not fully inserted.

APPLICABILITY: MODES 3*,** 4*,** and 5*,**

ACTION:

With less than the above required group demand position indicators(s) OPERABLE, immediately open the reactor trip system breakers.

SURVEILLANCE REQUIREMENTS

4.1.3.3.1 Each of the above required group demand rod position indicator(s) shall be determined to be OPERABLE by movement of the associate control rod at least 10 steps in any one direction at least once per 31 days.

4.1.3.3.2 A CHANNEL CHECK, CHANNEL CALIBRATION, and ANOLOG OPERATIONAL TEST shall be performed per Table 4.1-1.

* With the reactor trip system breakers in the closed position.
** See Special Test Exception 3.10.5.

TABLE 4.1-1

ROD POSITION INDICATION SURVEILLANCE REQUIREMENTS

Functional Unit	Channel Check	Channel Calibration	Analog Channel Operational Test
Individual Rod Position	S*	R	N*
Demand Position	S*	N/A	N/A

*Not required during Cold Shutdown

REACTIVITY CONTROL SYSTEMS

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full-length (shutdown and control) rod drop time from the fully withdrawn position shall be less than or equal to 2.4 seconds from beginning of rod motion to dashpot entry with:

a. T_{avg} greater than or equal to 541 F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2

ACTION:

a. With the drop time of any full-length rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to proceeding to MODE 1 or 2.

SURVEILLANCE REQUIREMENTS

4.1.3.4 The rod drop time of full-length rods shall be demonstrated through measurement prior to reactor criticality:

a. For all rods following each removal of the reactor vessel head,

b. For specifically affected individual rods following any maintenance on or modification to the Control Rod Drive System which could affect the drop time of those specific rods, and

c. At least once per refueling.

SHUTDOWN ROD INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown rods shall be fully withdrawn.

APPLICABILITY: MODES 1* and 2* **

ACTION:

With a maximum of one shutdown rod not fully withdrawn, except for surveillance testing pursuant to Specification 4.1.3.1.2, within 1 hour either:

a. Fully withdraw the rod, or

b. Declare the rod to be inoperable and apply Specification 3.1.3.1.

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each shutdown rod shall be determined to be fully withdrawn by use of the group demand counters, and verified by the analog rod position indicators within one hour after rod motion.

a. Within 15 minutes prior to withdrawal of any rods in control banks A, B, C or D during an approach to reactor criticality, and

b. At least once per 12 hours thereafter.

* See Special Test Exceptions Specifications 3.10.2 and 3.10.3

** With k_{eff} greater than or equal to 1.

REACTIVITY CONTROL SYSTEMS

CONTROL ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The control banks shall be limited in physical insertion as shown in Figure 3.1-1.

APPLICABILITY: MODES 1* and 2* **

ACTION:

With the control banks inserted beyond the above insertion limits, except for surveillance testing pursuant to Specification 4.1.3.1.2:

a. Restore the control banks to within the limits within 2 hours, or

b. Reduce THERMAL POWER within 2 hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the bank position using the above figures, or

c. Be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.6 The position of each control bank shall be determined to be within the insertion limits at least once per 12 hours, by use of the group demand counters and verified by the analog rod position indicators within one hour of rod motion, except during time intervals when the Rod Insertion Limit Monitor is inoperable, then verify the individual rod positions at least once per 4 hours.

* See Special Test Exceptions Specifications 3.10.2 and 3.10.3

** With k_{eff} greater than or equal to 1.

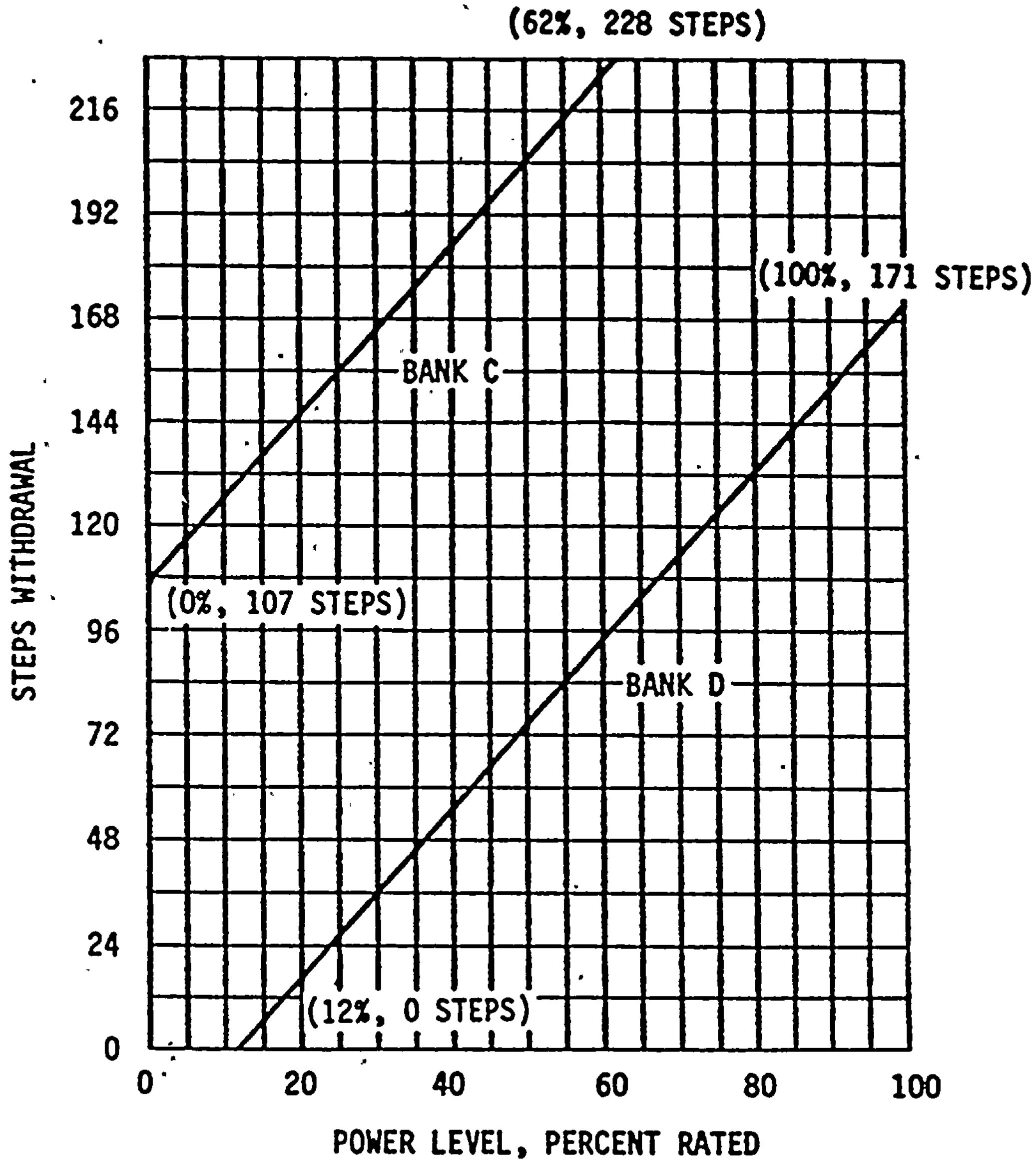

FIGURE 3.1-1

ROD BANK INSERTION LIMITS VERSUS THERMAL POWER

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within a ±5% target band (flux difference units) about the target flux difference.

The indicated AFD may deviate outside the above required target band at greater than or equal to 50% but less than 90% of RATED THERMAL POWER provided the indicated AFD is within the Acceptable Operation Limits of Figure 3.2-1 and the cumulative penalty deviation time does not exceed 1 hour during the previous 24 hours.

The indicated AFD may deviate outside the above required target band at greater than 15% but less than 50% of RATED THERMAL POWER provided the cumulative penalty deviation time does not exceed 1 hour during the previous 24 hours.

APPLICABILITY: MODE 1, above 15% of RATED THERMAL POWER.*

ACTION:

a. With the indicated AFD outside of the above required target band and with THERMAL POWER greater than or equal to 90% of RATED THERMAL POWER, within 15 minutes either:

   1. Restore the indicated AFD to within the target band limits, or

   2. Reduce THERMAL POWER to less than 90% of RATED THERMAL POWER.

b. With the indicated AFD outside of the above required target band for more than 1 hour of cumulative penalty deviation time during the previous 24 hours or outside the Acceptable Operation Limits of Figure 3.2-1 and with THERMAL POWER less than 90% but equal to or greater than 50% of RATED THERMAL POWER, reduce:

   1. THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes, and

   2. The Power Range Neutron Flux* ** - High Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

* See Special Test Exceptions Specification 3.10.2.

** Surveillance testing of the Power Range Neutron Flux Channel may be performed pursuant to Specification 4.3.1.1 provided the indicated AFD is maintained within the Acceptable Operation Limits of Figure 3.2-1. A total of 16 hours operation may be accumulated with the AFD outside of the above required target band during testing without penalty deviation.

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

c. With the indicated AFD outside of the above required target band for more than 1 hour of cumulative penalty deviation time during the previous 24 hours and with THERMAL POWER less than 50% but greater than 15% of RATED THERMAL POWER, the THERMAL POWER shall not be increased equal to or greater than 50% of RATED THERMAL POWER until the indicated AFD is within the above required target band.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The indicated AFD shall be determined to be within its limits during POWER OPERATION above 15% of RATED THERMAL POWER by:

a. Monitoring the indicated AFD for each OPERABLE excore channel:

1) At least once per 7 days when the alarm used to monitor the AFD is OPERABLE,and

2) At least once per hour for the first 24 hours after restoring the alarm used to monitor the AFD to OPERABLE status.

b. Monitoring and logging the indicated AFD for each OPERABLE excore channel at least once per hour for the first 24 hours and at least once per 30 minutes thereafter, when the alarm used to monitor the AFD is inoperable. The logged values of the indicated AFD shall be assumed to exist during the interval preceding each logging.

4.2.1.2 The indicated AFD shall be considered outside of its target band when two or more OPERABLE excore channels are indicating the AFD to be outside the target band. Penalty deviation outside of the above required target band shall be accumulated on a time basis of:

a. One minute penalty deviation for each 1 minute of POWER OPERATION outside of the target band at THERMAL POWER levels equal to or above 50% of RATED THERMAL POWER.

b. One-half minute penalty deviation for each 1 minute of POWER OPERATION outside of the target band at THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER.

4.2.1.3 The target flux difference of each OPERABLE excore channel shall be determined by measurement at least once per 92 Effective Full Power Days. The provisions of Specification 4.0.4 are not applicable.

4.2.1.4 The target flux difference shall be updated at least once per 31 Effective Full Power Days by either determining the target flux difference pursuant to Specification 4.2.1.3 above or by linear interpolation between the most recently measured value and predicted value at the end of cycle life. The provisions of Specification 4.0.4 are not applicable.

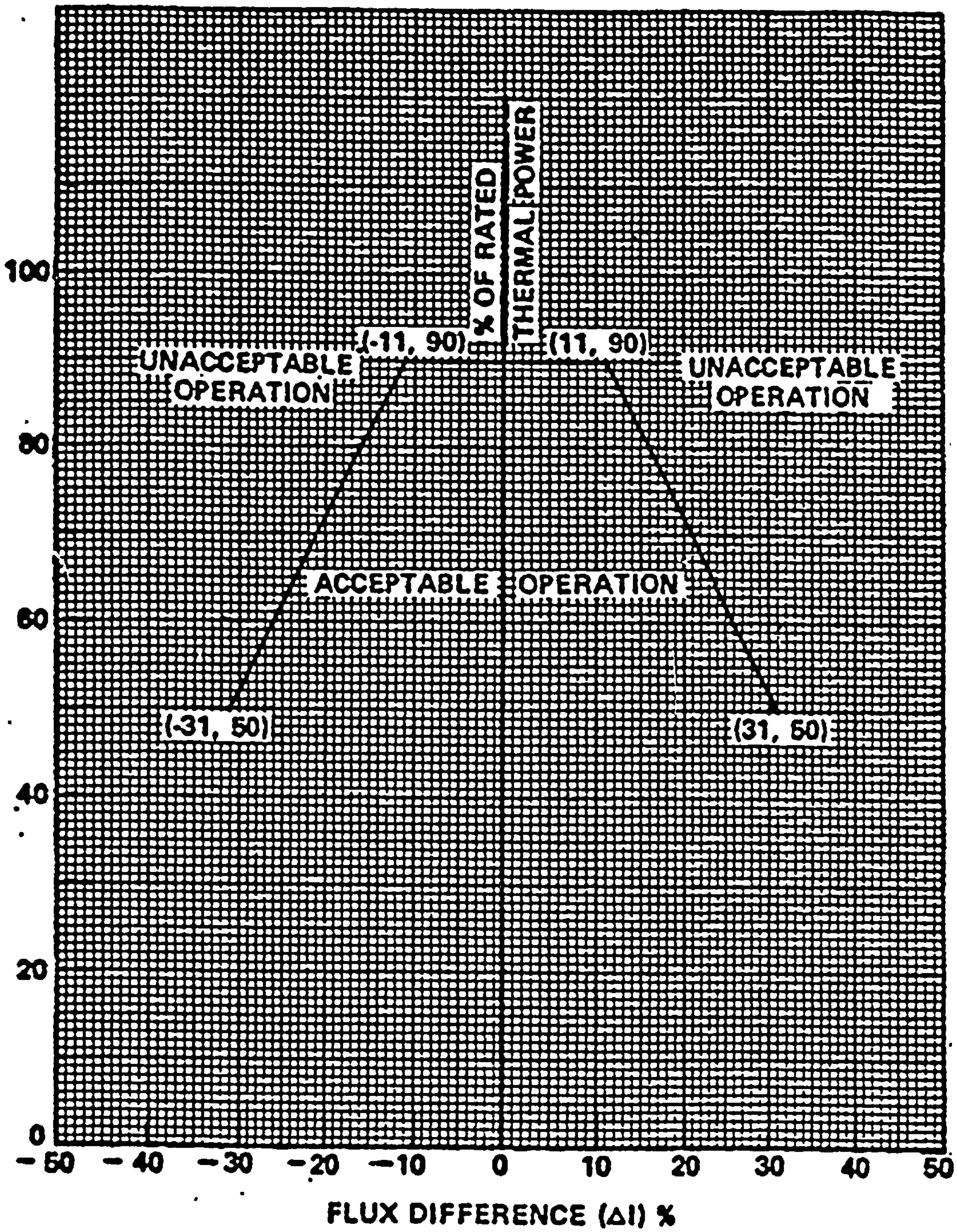

FIGURE 3.2-1

AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF
RATED THERMAL POWER

3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR - $F_Q(Z)$

LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q^L(Z)$ shall be limited by the following relationships:

$$F_Q^M(Z) \leq \frac{[F_Q]^L}{P} \times [K(Z)] \text{ for } P > 0.5$$

$$F_Q^M(Z) \leq \frac{[F_Q]^L}{0.5} \times [K(Z)] \text{ for } P \leq 0.5$$

where: $[F_Q]^L = 2.32$

$$P = \frac{\text{Thermal Power}}{\text{Rated Thermal Power}}$$

and K(Z) is the function obtained from Figure 3.2-2 for a given core height location.

APPLICABILITY: MODE 1

ACTION:

With the measured value of $F_Q(Z)$ exceeding its limit:

a. Reduce THERMAL POWER at least 1% for each 1% $F_Q^M(Z)$ exceeds $F_Q^L(Z)$ within 15 minutes and similarly reduce the Power Range Neutron Flux - High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower Delta-T Trip Setpoints (value of K_4) have been reduced at least 1% for each 1% $F_Q^M(Z)$ exceeds the limit; and

b. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced power limit required by ACTION a., above; THERMAL POWER may then be increased provided $F_Q^M(Z)$ is demonstrated through incore mapping to be within its limit.

SURVEILLANCE REQUIREMENTS

4.2.2.1 If $[F_Q]^P$ as predicted by approved physics calculations is greater than $[F_Q]^L$ and P is greater than P_T as defined in 4.2.2.2, $F_Q(Z)$ shall be evaluated by 4.2.2.2, 4.2.2.3 or 4.2.2.4 to determine if F_Q is within its limit. If $[F_Q]^P$, is less than $[F_Q]^L$ or P is less than P_T, $F_Q(Z)$ shall be evaluated to determine if $F_Q(Z)$ is within its limit as follows:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER.

b. Increasing the measured $F_Q(Z)$ component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties. Verifying that the requirements of Specification 3.2.2. are satisfied.

c. $F_Q^M(Z) \leq F_Q^L(Z)$

Where $F_Q^M(Z)$ is the measured $F_Q(Z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty and $F_Q^L(Z)$ is the F_Q limit defined in 3.2.2.

SURVEILLANCE REQUIREMENTS (Continued)

d. Measuring $F_Q^M(Z)$ according to the following schedule:

1. Prior to exceeding 75% of RATED THERMAL POWER*, after refueling,

2. At least once per 31 Effective Full Power Days.

e. With the relationship specified in Specification 4.2.2.1.c above not being satisfied:

1) Calculate the percent $F_Q^M(Z)$ exceeds its limit by the following expression:

$$\left[\begin{matrix}\text{Maximum} \\ \text{Over Z}\end{matrix} \left[\frac{F_Q^M(Z)}{[F_Q]^L \times K(Z)/P}\right] - 1\right] \times 100 \text{ for } P \geq 0.5$$

$$\left[\begin{matrix}\text{Maximum} \\ \text{Over Z}\end{matrix} \left[\frac{F_Q^M(Z)}{[F_Q]^L \times K(Z)/0.5}\right] - 1\right] \times 100 \text{ for } P < 0.5$$

* During power escalation at the beginning of each cycle, power level may be increased until a power level for extended operation has been achieved and power distribution map obtained.

SURVEILLANCE REQUIREMENTS (Continued)

2) The following action shall be taken:

a) Comply with the requirements of Specification 3.2.2 for $F_Q^M(Z)$ exceeding its limit by the percent calculated above.

4.2.2.2 Operation is permitted at power above P_T where P_T equals the ratio of $[F_Q]^L$ divided by $[F_Q]^P$ if the following Augmented Surveillance (Movable Incore Detection System, MIDS) requirements are satisfied:

a. The axial power distribution shall be measured by MIDS when required such that the limit of $[F_Q]^L/P$ times Figure 3.2.2 is not exceeded. $F_j(Z)$ is the normalized axial power distribution from thimble j at core elevation (Z).

1. If $F_j(Z)$ exceeds $[F_j(Z)]_S$ as defined in the bases by $\leq$ 4%, immediately reduce thermal power one percent for every percent by which $[F_j(Z)]_S$ is exceeded.

2. If $F_j(Z)$ exceeds $[F_j(Z)]_S$ by > 4% immediately reduce thermal power below P_T. Corrective action to reduce $F_j(Z)$ below the limit will permit return to thermal power not to exceed current P_L as defined in the bases.

SURVEILLANCE REQUIREMENTS (Continued)

b. $F_j(Z)$ shall be determined to be within limits by using MIDS to monitor the thimbles required per specification 4.2.2.2.c at the following frequencies.

1. At least once every 24 hours, and

2. Immediately following and as a minimum at 2, 4 and 8 hours following the events listed below and every 24 hours thereafter.

   1) Raising the thermal power above P_T, or

   2) Movement of control-bank D more than an accumulated total of 15 steps in any one direction.

c. MIDS shall be operable when the thermal power exceeds P_T with:

1. At least two thimbles available for which $\bar{R}_j$ and σ_j as defined in the bases have been determined.

2. At least two movable detectors available for mapping $F_j(Z)$.

3. The continued accuracy and representativeness of the selected thimbles shall be verified by using the most recent flux map to update the $\bar{R}$ for each selected thimble. The flux map must be updated at least once per 31 effective full power days.

SURVEILLANCE REQUIREMENTS (Continued)

4.2.2.3 Base Load operation is permitted at powers above P_T if the following requirements are satisfied:

a. Either of the following preconditions for Base Load operation must be satisfied.

1. For entering Base Load operation with power less than P_T,

a) Maintain THERMAL POWER between $P_T/1.05$ and P_T for at least 24 hours,

b) Maintain the AFD (Delta-I) to within a ± 2% or ± 3% target band for at least 23 hours per 24 hour period.

c) After 24 hours have elapsed, take a full core flux map to determine $F_Q^M(Z)$ unless a valid full core flux map was taken within the time period specified in 4.2.2.1d.

d) Calculate P_{BL} per 4.2.2.3b.

2. For entering Base Load operation with power greater than P_T,

a) Maintain THERMAL POWER between P_T and the power limit determined in 4.2.2.2 for at least 24 hours, and maintain Augmented Surveillance requirements of 4.2.2.2 during this period.

b) Maintain the AFD (Delta-I) to within a ± 2% or ± 3% target band for at least 23 hours per 24 hour period,

c) After 24 hours have elapsed, take a full core flux map to determine $F_Q^M(Z)$ unless a valid full core flux map was taken within the time period specified in 4.2.2.1d.

d) Calculate P_{BL} per 4.2.2.3b.

b. Base Load operation is permitted provided:

1. THERMAL POWER is maintained between P_T and P_{BL} or between P_T and 100% (whichever is most limiting).

2. AFD (Delta-I) is maintained within a ± 2% or ± 3% target band.

3. Full core flux maps are taken at least once per 31 effective Full Power Days.

P_{BL} and P_T are defined as:

$$P_{BL} = \underset{\text{Over } Z}{\text{Minimum}} \left[\frac{[F_Q]^L \times K(Z)}{F_Q^M(Z) \times W(Z)_{BL} \times 1.09} \right]$$

$$P_T = [F_Q]^L / [F_Q]^P$$

where: $F_Q^M(Z)$ is the measured $F_Q(Z)$ with no allowance for manufacturing tolerances or measurement uncertainty. For the purpose of this Specification $[F_Q(Z)]_{Map}$ Meas. shall be obtained between elevations bounded by 10% and 90% of the active core height. $[F_Q]^L$ is the F_Q limit. K(Z) is given in Figure 3.2-2. $W(Z)_{BL}$ is the cycle dependent function that accounts for limited power distribution transients encountered during base load operation.

The function is given in the Peaking Factor Limit Report as per Specification 6.9.1.6. The 9% uncertainty factor accounts for manufacturing tolerance, measurement error, rod bow and any burnup and power dependent peaking factor increases.

c. During Base Load operation, if the THERMAL POWER is decreased below P_T, then the conditions of 4.2.2.3.a shall be satisfied before re-entering Base Load operation.

d. If any of the conditions of 4.2.2.3b are not maintained, reduce THERMAL POWER to less than or equal to P_T, or, within 15 minutes initiate the Augmented Surveillance (MIDS) requirements of 4.2.2.2.

4.2.2.4 Operation is permitted at powers above P_T if the following Radial Burndown conditions are satisfied:

a. Radial Burndown operation is restricted to use at powers between P_T and P_{RB} or P_T and 1.00 (whichever is most limiting). The maximum relative power permitted under Radial Burndown operation, P_{RB}, is equal to the minimum value of the ratio of $[F_Q^L(Z)]/[F_Q(Z)]_{RB}$ Meas. where:

$[F_Q(Z)]_{RB}$ Meas. = $[F_{xy}(Z)]_{Map}$ Meas. x $F_z(Z)$ x 1.09 and

$[F_Q^L(Z)]$ is equal to $[F_Q^L]$ x K(Z).

b. A full core flux map to determine $[F_{xy}(Z)]_{Map}$ Meas. shall be taken within the time period specified in Section 4.2.2.1d.2. For the purpose of the specification, $[F_{xy}(Z)]_{Map}$ Meas. shall be obtained between the elevations bounded by 10% and 90% of the active core height.

SURVEILLANCE REQUIREMENTS (Continued)

c. The function $F_Z(Z)$, provided in the Peaking Factor Limit Report (6.9.1.6), is determined analytically and accounts for the most perturbed axial power shapes which can occur under axial power distribution control. The uncertainty factor of 9% accounts for manufacturing tolerances, measurement error, rod bow, and any burnup dependent peaking factor increases.

d. Radial Burndown operation may be utilized at powers between P_T and P_{RB}, or, P_T and 1.00 (whichever is most limiting) provided that the AFD (Delta-I) is within ± 5% of the target axial offset.

e. If the requirements of Section 4.2.2.4d are not maintained, then the power shall be reduced to less than or equal to P_T, or within 15 minutes Augmented Surveillance of hot channel factors shall be initiated if the power is above P_T.

4.2.2.5 When $F_Q(Z)$ is measured for reasons other than meeting the requirements of specification 4.2.2.1, 4.2.2.2, 4.2.2.3 or 4.2.2.4 an overall measured $F_Q(Z)$ shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

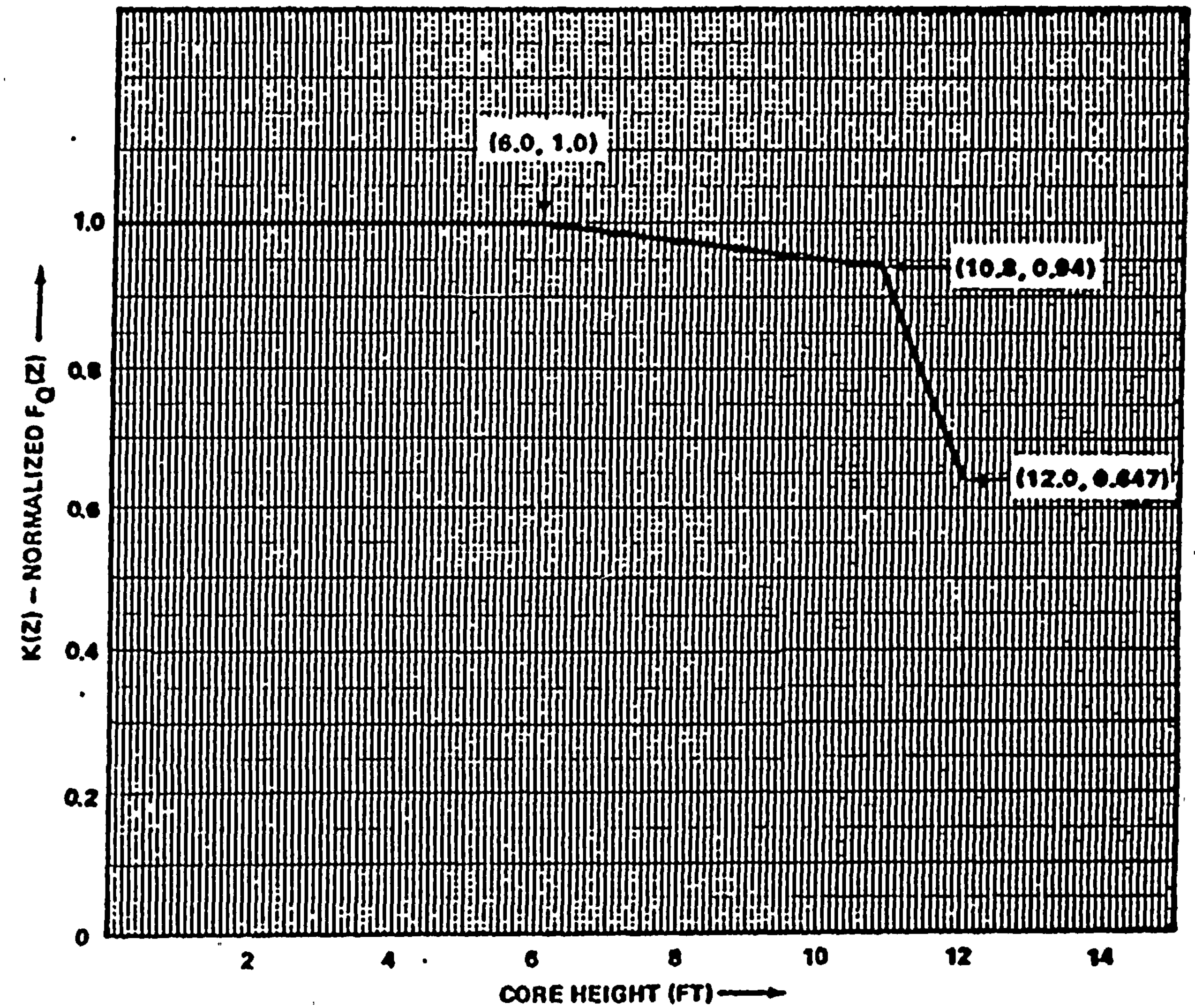

FIGURE 3.2-2
K(Z) Normalized $F_Q(Z)$ as a Function of Core Height

3/4.2.3 NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

LIMITING CONDITION FOR OPERATION

3.2.3 $F^N_{\Delta H}$ Shall be limited to the following:

$F^N_{\Delta H} \leq 1.62\ [1.0 + 0.3(1-P)]$, where

$$P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$$

APPLICABILITY: MODE 1.

ACTION:

With $F^N_{\Delta H}$ exceeding its limit:

a. Within 2 hours reduce THERMAL POWER to less than 50% of RATED THERMAL POWER and reduce the Power Range Neutron Flux - High Trip Setpoint to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

b. Within 24 hours of initially being outside the above limits, verify through incore flux mapping that $F^N_{\Delta H}$ is restored to within the above limit, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION a. and/or b., above; subsequent POWER OPERATION may proceed provided that $F^N_{\Delta H}$ is demonstrated, through incore flux mapping, to be within its limit of acceptable operation prior to exceeding the following THERMAL POWER levels:

1. A nominal 50% of RATED THERMAL POWER,
2. A nominal 75% of RATED THERMAL POWER, and
3. Within 24 hours of attaining greater than or equal to 95% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 $F^N_{\Delta H}$ shall be determined to be within its limit through incore flux mapping:

a. Prior to operating above 75% of RATED THERMAL POWER after each fuel loading, and

b. At least once per 31 Effective Full Power Days.

4.2.3.3 The measured $F^N_{\Delta H}$ shall be increased by 4% to account for measurement error.

POWER DISTRIBUTION LIMITS

3/4.2.4 QUADRANT POWER TILT RATIO

LIMITING CONDITION FOR OPERATION

3.2.4 The QUADRANT POWER TILT RATIO shall not exceed 1.02.

APPLICABILITY: MODE 1, above 50% of RATED THERMAL POWER*

ACTION:

a. With the QUADRANT POWER TILT RATIO determined to exceed 1.02 but less than or equal to 1.09:

   1. Calculate the QUADRANT POWER TILT RATIO at least once per hour until either:

      a) The QUADRANT POWER TILT RATIO is reduced to within its limit or

      b) THERMAL POWER is reduced to less than 50% of RATED THERMAL POWER.

   2. Within 2 hours either:

      a) Reduce the QUADRANT POWER TILT RATIO to within its limit, or

      b) Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1 and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours.

   3. Verify that the QUADRANT POWER TILT RATIO is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours; and

   4. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

---

* See Special Test Exceptions Specification 3.10.2.

ACTION (Continued)

b. With the QUADRANT POWER TILT RATIO determined to exceed 1.09 due to misalignment of either a shutdown or control rod:

1. Calculate the QUADRANT POWER TILT RATIO at least once per hour until either:

a) The QUADRANT POWER TILT RATIO is reduced to within its limit, or

b) THERMAL POWER is reduced to less than 50% of RATED THERMAL POWER.

2. Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1, within 30 minutes;

3. Verify that the QUADRANT POWER TILT RATIO is within its limit within 2 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours; and

4. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

c. With the QUADRANT POWER TILT RATIO determined to exceed 1.09 due to causes other than the misalignment of either a shutdown or control rod:

1. Calculate the QUADRANT POWER TILT RATIO at least once per hour until either:

a) The QUADRANT POWER TILT RATIO is reduced to within its limit, or

b) THERMAL POWER is reduced to less than 50% of RATED THERMAL POWER.

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoint to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours; and

3. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified at 95% or greater RATED THERMAL POWER.

d. The provisions of Specifications 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.2.4.1 The QUADRANT POWER TILT RATIO shall be determined to be within the limit above 50% of RATED THERMAL POWER by:

a. Calculating the ratio at least once per 7 days when the Power Range Upper Detector High Flux Deviation and Power Range Lower Detector Flux Deviation Alarms are OPERABLE, and

b. Calculating the ratio at least once per 12 hours during steady-state operation when either alarm is inoperable.

4.2.4.2 The QUADRANT POWER TILT RATIO shall be determined to be within the limit when above 75% of RATED THERMAL POWER with one Power Range channel inoperable by using the movable incore detectors to confirm that the normalized symmetric power distribution, obtained from two sets of four symmetric thimble locations or full-core flux map, is consistent with the indicated QUADRANT POWER TILT RATIO at least once per 12 hours.

4.2.4.3 If the QUADRANT POWER TILT RATIO is not within its limit within 24 hours and the POWER DISTRIBUTION LIMITS of 3.2.2 and 3.2.3 are within their limits, a Special Report as identified in 6.9.2 shall be submitted within 30 days including an evaluation of the cause of the discrepancy.

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB-related parameters shall be maintained within the limits shown on Table 3.2-1:

a. Reactor Coolant System T_{avg}

b. Pressurizer Pressure, and

c. Reactor Coolant Flow.

APPLICABILITY: MODE 1.

ACTION:

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the parameters of Table 3.2-1 shall be verified to be within its limits at least once per 12 hours.

4.2.5.2 The RCS flow rate indicators shall be subjected to a CHANNEL CALIBRATION at least once per refueling.

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	
Indicated Reactor Coolant System T_{avg}	$\leq$ 576.3 F
Indicated Pressurizer Pressure	$\geq$ 2237 psia*
Indicated Reactor Coolant Flow	$\geq$ 277,900 gpm

* Limit not applicable during either a THERMAL POWER ramp in excess of 5% of RATED THERMAL POWER per minute or a THERMAL POWER step in excess of 10% of RATED THERMAL POWER.

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the Reactor Trip System instrumentation channels and interlocks of Table 3.3-1 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each Reactor Trip System instrumentation channel and interlock and the automatic trip logic shall be demonstrated OPERABLE by the performance of the Reactor Trip System Instrumentation Surveillance Requirements specified in Table 4.3-1.

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1,2	1
	2	1	2	3*,4*,5*	9
2. Power Range, Neutron Flux -					
High Setpoint	4	2	3	1,2	2**
Low Setpoint	4	2	3	1****,2	2**
3. Intermediate Range, Neutron Flux	2	1	2	1****,2	3
4. Source Range, Neutron Flux					
A. Startup	2	1	2	2***	4
B. Shutdown	2	1	2	3*,4*,5*	9
C. Shutdown	2	0	1	3,4,5	5
5. Overtemperature ΔT	3	2	2	1,2	6**
6. Overpower ΔT	3	2	2	1,2	6**
7. Pressurizer Pressure-Low	3	2	2	1,2	6**
8. Pressurizer Pressure-High	3	2	2	1,2	6**
9. Pressurizer Water Level-High	3	2	2	1,2	6**
10. Loss of Flow					
A. Single Loop (Above P-8)	3/loop	2/loop	2/loop	1,2	6**
B. Two Loops (Above P-7 and below P-8)	3/loop	2/loop	2/loop	1,2	6**
11. Steam Generator Water Level-Low-Low	3/stm. gen.	2/stm. gen.	2/stm. gen.	1,2	6**

TABLE 3.3-1

(Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
12. Steam Generator Water Level-Low Coincident With Steam/Feedwater Flow Mismatch	2 stm. gen. level and 2 stm/feed-flow mis-match in each stm. gen.	1 stm.gen. level coincident with 1 stm/feed-flow mismatch in same stm. gen.	2 stm. gen. level or 2/stm/feed-flow mis-match in each stm. gen.	1,2	10**
13. Undervoltage-A and B 4KV Bus	2/bus	1/bus on both buses	2/bus	1,2	11**
14. Underfrequency Trip of Reactor Coolant Pump (not a direct trip of reactor)	2/bus	1 to trip RCP(s)	1/bus	1,2	11**
15. Turbine Trip					
A. Autostop Oil Pressure	3	2	2	1,2	11**
B. Turbine Stop Valve Closure	2	2	2	1,2	11**
16. Safety Injection Input from ESF	2	1	2	1,2	8
17. Reactor Coolant Pump Breaker Position Trip					
A. Above P-8	1/breaker	1	1/breaker	1,2	11**
B. Above P-7 and below P-8	1/breaker	2	1/breaker per operating loop	1,2	11**

TABLE 3.3-1
(Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
18. Reactor Trip System Interlocks					
A. Intermediate Range Neutron Flux, P-6	2	1	1	2***	7
B. Low Power Reactor Trips Block, P-7, P-10 Input or	4	2	3	1	7
Turbine First Stage Pressure	2	1	1	1	7
C. Power Range Neutron Flux,P-8	4	2	3	1	7
D. Low Setpoint Power Range Neutron Flux, P-10	4	2	3	1,2	7
19. Reactor Trip Breakers	2	1	2	1,2	8
	2	1	2	3*,4*,5*	9
20. Automatic Trip Logic	2	1	2	1,2	8
	2	1	2	3*,4*,5*	9

TABLE 3.3-1
(Continued)

TABLE NOTATION

* With the reactor trip system breakers in the closed position, the control rod drive system capable of rod withdrawal.

** The provisions of Specification 3.0.4 are not applicable.

*** Below the P-6 (Intermediate Range Neutron Flux Interlock) setpoint.

**** Below the P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.

ACTION STATEMENTS

ACTION 1 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT STANDBY within the next 6 hours.

ACTION 2 - With the number of OPERABLE channels one less than the Total number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 1 hour.

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 2 hours for surveillance testing of other channels per Specification 4.3.1.1.

c. Either THERMAL POWER is restricted to less than or equal to 75% of RATED THERMAL POWER and the Power Range Neutron Flux trip setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER within 4 hours; or, the QUADRANT POWER TILT RATIO is monitored at least once per 12 hours per Specification 4.2.4.2.

ACTION 3 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement and with the THERMAL POWER level:

a. Below the P-6 (Intermediate Range Neutron Flux Interlock) setpoint, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the P-6 Setpoint.

b. Above the P-6 (Intermediate Range Neutron Flux Interlock) setpoint but below 10 % of RATED THERMAL POWER, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 10 % of RATED THERMAL POWER.

ACTION 4 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement suspend all operations involving positive reactivity changes.

TABLE 3.3-1
(Continued)

ACTION STATEMENTS

ACTION 5 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.1 or 3.1.1.2, as applicable, within 1 hour and at least once per 12 hours thereafter.

ACTION 6 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed until performance of the next required OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 1 hour.

ACTION 7 - With less than the Minimum Number of Channels OPERABLE, within one hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

ACTION 8 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is OPERABLE. One channel maybe out of service for up to 8 hours for relay replacement.

ACTION 9 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the reactor trip breakers within the next hour.

ACTION 10 - With the number of OPERABLE channels less than the total number of channels, STARTUP and/or POWER OPERATION may proceed until performance of the next required OPERATIONAL TEST provided the inoperable channels are placed in the tripped condition within 1 hour.

ACTION 11 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours.

TABLE 4.3-1
REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

	FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1.	Manual Reactor Trip	N.A.	N.A.	N.A.	R	N.A	1,2,3*,4*,5*
2.	Power Range, Neutron Flux						
	a. High Setpoint	S	D(2,4), M(3,4), Q(4,6), R(4,5),	M	N.A.	N.A	1.2
	b. Low Setpoint	S	R(4)	M	N.A.	N.A.	1***,2
3.	Intermediate Range,	S	R(4,5)	S/U(1),M	N.A.	N.A.	1***,2
4.	Source Range, Neutron Flux	S	R(4,5)	S/U(1),M(9)	N.A.	N.A.	2**,3,4,5
5.	Overtemperature ΔT	S	R(12)	M	N.A.	N.A.	1,2
6.	Overpower ΔT	S	R	M	N.A.	N.A.	1,2
7.	Pressurizer Pressure--Low	S	R	M	N.A.	N.A.	1,2
8.	Pressurizer Pressure--High	S	R	M	N.A.	N.A.	1,2
9.	Pressurizer Water Level--High	S	R	M	N.A.	N.A.	1,2
10.	Reactor Coolant Flow--Low	S	R	M	N.A.	N.A.	1,2
11.	Steam Generator Water Level--Low-Low	S	R	M	N.A.	N.A.	1,2
12.	Steam Generator Water Level--Low Coincident with Steam/Feedwater Flow Mismatch	S	R	M	N.A.	N.A.	1,2

TABLE 4.3-1
(Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
13. Undervoltage - A and B 4KV Bus	N.A.	R	N.A.	Part of Channel Calibration	N.A.	1,2
14. Underfrequency - Trip of Reactor Coolant Pumps	N.A.	R	N.A.	Part of Channel Calibration	N.A.	1,2
15. Turbine Trip						
a. Autostop Oil Pressure	N.A.	R	N.A.	S/U(1,10)	N.A.	1,2
b. Turbine Stop Valve Closure	N.A.	R	N.A.	S/U(1,10)	N.A.	1,2
16. Safety Injection Input from ESF	N.A.	N.A.	N.A.	R	N.A.	1,2
17. Reactor Trip System Interlocks						
a. Intermediate Range Neutron Flux, P-6	N.A.	R(4)	M	N.A.	N.A.	2**
b. Low Power Reactor Trips Block, P-7	N.A.	R(4)	M(8)	N.A.	N.A.	1
c. Power Range Neutron Flux, P-8	N.A.	R(4)	M(8)	N.A.	N.A.	1
d. Power Range Neutron Flux, P-10	N.A.	R(4)	M(8)	N.A.	N.A.	1,2
e. Turbine First Stage Pressure	N.A.	R	M(8)	N.A.	N.A.	1
18. Reactor Trip Breaker	N.A.	N.A.	N.A.	M(7,11)	N.A.	1,2,3*,4*,5*
19. Automatic Trip Logic	N.A.	N.A.	N.A.	N.A	M(7)	1,2,3*,4*,5*

TABLE 4.3-1

(Continued)
TABLE NOTATIONS

*	Only with the Reactor Trip System breakers closed and the Control Rod Drive System capable of rod withdrawal.
**	Below P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.
***	Below P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.

(1) If not performed in previous 7 days

(2) Comparison of calorimetric to excore power indication above 15% of RATED THERMAL POWER. Adjust excore channel gains consistent with calorimetric power if absolute difference is greater than 2%. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(3) Single point comparison of incore to excore AXIAL FLUX DIFFERENCE above 15% of RATED THERMAL POWER. Recalibrate if the absolute difference is greater than or equal to 3%. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(4) Neutron detectors may be excluded from CHANNEL CALIBRATION.

(5) Detector plateau curves shall be obtained, and evaluated and compared to manufacturer's data. For the Intermediate Range and Power Range Neutron Flux channels the provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(6) Incore - Excore Calibration, above 75% of RATED THERMAL POWER. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(7) Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.

(8) With power greater than or equal to the Interlock Setpoint the required ANALOG CHANNEL OPERATIONAL TEST shall consist of verifying that the interlock is in the required state by observing the permissive annunciator window.

(9) Monthly surveillance in MODES 3*, 4*, and 5* shall also include verification that permissives P-6 and P-10 are in their required state for existing plant conditions by observation of the permissive annunciator window.

(10) Setpoint verification is not applicable.

TABLE 4.3-1
(Continued)

TABLE NOTATIONS

(11) At least once per 18 months and following maintenance or adjustment of the Reactor trip breakers, the TRIP ACTUATING DEVICE OPERATIONAL TEST shall include independent verification of the Undervoltage and Shunt trips.

(12) CHANNEL CALIBRATION shall include the RTD bypass loops flow rate.

INSTRUMENTATION

3/4.3.2 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2 The Engineered Safety Features Actuation System (ESFAS) instrumentation channels and interlocks shown in Table 3.3-2 shall be OPERABLE with their Trip Setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-3.

APPLICABILITY: As shown in Table 3.3-2.

ACTION:

a. With an ESFAS Instrumentation or Interlock Trip Setpoint less conservative than the value shown in the Trip Setpoint column but more conservative than the value shown in the Allowable Value column of Table 3.3-3, adjust the Setpoint consistent with the Trip Setpoint value.

b. With an ESFAS instrumentation or interlock Trip Setpoint less conservative than the value shown in the Allowable Value Column of Table 3.3-3, declare the channel inoperable and apply the applicable ACTION statement requirements of Table 3.3-2 until the channel is restored to OPERABLE status with its Setpoint adjusted consistent with the Trip Setpoint value.

c. With an ESFAS instrumentation channel or interlock inoperable, take the ACTION shown in Table 3.3-2.

SURVEILLANCE REQUIREMENTS

4.3.2.1 Each ESFAS instrumentation channel and interlock and the automatic actuation logic and relays shall be demonstrated OPERABLE by performance of the ESFAS Instrumentation Surveillance Requirements specified in Table 4.3-2.

TABLE 3.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

	FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1.	Safety Injection					
	a. Manual Initiation	2	1	2	1, 2, 3, 4	17
	b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
	c. High Containment Pressure	3	2	2	1, 2, 3	15*
	d. Pressurizer Low Pressure	3	2	2	1, 2, 3**	15*
	e. High Differential Pressure Between the Steam Line Header and any Steam Line	3/steam line	2/steam line in any steam line	2/steam line	1, 2, 3***	15*
	f. High Steam Line Flow	2/steam line	1/steam line in any two steam lines	1/steam line in any two steam lines	1, 2, 3***	15*
	Coincident with:					
	Low Steam Line Pressure or	1/steam line	1/steam line in any two steam lines	1/steam line in any two steam lines	1, 2, 3***	15*
	Low T_{avg}	1/loop	1/loop in any two loops	1/loop in any two loops	1, 2, 3***	15*

TABLE 3.3-2
(Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
2. Containment Spray					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
b. Containment Pressure--High-High	3	2	2	1, 2, 3, 4	15
Coincident with:					
High Containment Pressure	3	2	2	1, 2, 3, 4	15*
3. Containment Isolation					
a. Phase "A" Isolation					
1) Manual Initiation	2	1	2	1, 2, 3, 4	17
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
3) Safety Injection	See Item 1. above for all automatic Safety Injection initiating functions and requirements. (Manual S.I. initiation will not initiate Phase A Isolation.)				

TABLE 3.3-2
(Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
b. Phase "B" Isolation					
1) Manual Initiation	2	2 (Both buttons must be pushed)	2	1, 2, 3, 4	17
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
3) Containment Pressure-High-High Coincident with High Containment Pressure	See 2.b. above for instrumentation requirements.				
c. Containment Ventilation Isolation					
1) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	16
2) Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
3) High Containment Radioactivity	2 (R11 and R12)	1	2	1, 2, 3, 4	16, 16A

TABLE 3.3-2
(Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
4) Manual Phase A or Phase B Containment Isolation	2	1	2	1, 2, 3, 4	17
4. Steam Line Isolation					
a. Manual Initiation	1/steam line	1/steam line	1/steam line	1, 2, 3	21
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	20
c. Containment Pressure--High-High Coincident with High Containment Pressure	See Item 2.b. above for instrumentation requirements.				
d. High Steam Line Flow Coincident with Low Steam Line Pressure or Low T_{avg}	See Item 1.f. above for instrumentation requirements.				

TABLE 3.3-2
(Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
5. Feedwater Isolation					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2	22
b. Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements				
6. Auxiliary Feedwater					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	20
b. Stm. Gen. Water Level--Low-Low	3/Steam Generator	2/Steam Generator in any S.G	2/steam Generator	1, 2, 3	15*
c. Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				

TABLE 3.3-2
(Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS TO TRIP	CHANNELS OPERABLE	APPLICABLE MODES	ACTION
d. Loss of Power					
1) 4.16 KV Bus Loss of Voltage	2/Bus	2/Bus	1/Bus	1, 2, 3	18
2) 480V Load Centers	2 Instantaneous Relays per Load Center	2 on any Load Center	1/Load Center	1, 2, 3	18
3) 480V Load Centers	2 Inverse Time Delay Relays per Load Center	2 on any Load Center	1/Load Center	1, 2, 3	18
e. Trip of All Main Feedwater Pump Breakers	1/Breaker	1/Breaker/	1/Breaker/ operating pump	1, 2, 3 operating pump	23
7. Loss of Power 4.16 KV Bus or 480V Load Centers	2/Bus	2/Bus	1/Bus	1, 2, 3, 4	18
8. Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure, P-11	3	2	2	1, 2, 3	19

TABLE 3.3-2
(Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS TO TRIP	CHANNELS OPERABLE	APPLICABLE MODES	ACTION
b. Low T_{avg}	3	2	3	1, 2, 3	19

TABLE 3.3-2 (Continued)

TABLE NOTATIONS

* The provisions of Specification 3.0.4 are not applicable.

** Trip function may be blocked in this MODE below the Pressurizer Pressure Interlock Setpoint of 2000 psig.

*** Trip function may be blocked in this MODE below the Low T_{avg} Interlock Setpoint.

ACTION STATEMENTS

ACTION 14 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1, provided the other channel is OPERABLE.

ACTION 15 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed until performance of the next required ANALOG CHANNEL OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 1 hour.

ACTION 16 - With less than the Minimum Channels OPERABLE requirement, operation may continue provided the containment purge supply and exhaust valves are maintained closed.

ACTION 16A - Comply with ACTION requirements of Specification 3.4.6.1.

ACTION 17 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 18 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the inoperable channel is placed in the tripped condition within 1 hour.

TABLE 3.3-2 (Continued)

TABLE NOTATIONS

ACTION STATEMENTS (Continued)

ACTION 19 - With less than the Minimum Number of Channels OPERABLE, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

ACTION 20 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

ACTION 21 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or declare the associated valve inoperable and take the ACTION required by Specification 3.7.1.5.

ACTION 22 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

ACTION 23 - With the number of OPERABLE channels one less than the Minimum Channel OPERABLE requirement, comply with Specification 3.0.3.

TABLE 3.3-3

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
1. Safety Injection		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic	N.A.	N.A.
c. High Containment Pressure	$\leq$ 4 psig	$\leq$ 4.5 psig
d. Pressurizer Low Pressure	$\geq$ 1723 psig	$\geq$ 1713 psig
e. High Differential Pressure between the Steam Line Header and any Steam Line	$\leq$ 150 psi	$\leq$ 162 psi
f. High Steam Line Flow	$\leq$ A function defined as follows: A dp corresponding to 0.64 $\times 10^6$ lbs/hr at 0% load increasing linearly to a dp corresponding to 3.84 $\times 10^6$ lbs/hr. at full load	$\leq$ A function defined as follows: A dp corresponding to 0.76 $\times 10^6$ lbs/hr at 0% load increasing linearly to a dp corresponding to 3.96$\times 10^6$ lbs/hr at full load
Coincident with:		
Low Steam Line Pressure or	$\geq$ 600 psig	$\geq$ 580 psig
Low T_{avg}	$\geq$ 533 F	$\geq$ 531 F
2. Containment Spray		
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
b. Containment Pressure--High-High	$\leq$ 30 psig	$\leq$ 32 psig
Coincident with:		
High Containment Pressure	$\leq$ 4 psig	$\leq$ 4.5 psig

TABLE 3.3-3
(Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
3. Containment Isolation		
a. Phase "A" Isolation		
1) Manual Initiation	N.A.	N.A.
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
3) Safety Injection	See Item 1 above for Safety Injection Trip Setpoints and Allowable Values	
b. Phase "B" Isolation		
1) Manual Initiation	N.A.	N.A.
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
3) Containment Pressure--High-High Coincident with High Containment Pressure	See Item 2.b. above for Trip Setpoints and Allowable Values	
c. Containment Ventilation Isolation		
1) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
2) Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values	
3) High Containment Radioactivity	See Specification 3.3.3.1 for Setpoints for R-11 and R-12	
4) Manual Phase A or Phase B Containment Isolation	N.A.	N.A.

TABLE 3.3-3
(Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
4. Steam Line Isolation		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Containment Pressure--High-High Coincident with High Containment Pressure	See Item 2.b. above for Trip Setpoints and Allowable Values	
d. High Steam Flow Coincident with Low Steam Line Pressure or Low T_{avg}	See Item 1.f. above for Trip Setpoints and Allowable Values	
5. Feedwater Isolation		
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
b. Safety Injection	See Item 1. above for all Trip Setpoints and Allowable Values	
6. Auxiliary Feedwater		
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
b. Steam Generator Water Level--Low-Low	$\geq$ 15% of narrow range instrument span	$\geq$ 14% of narrow range instrument span
c. Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values	

TABLE 3.3-3
(Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT		TRIP SETPOINT	ALLOWABLE VALUE
6. Auxiliary Feedwater (Continued).			
d. Loss of Power			
1) 4.16 KV Bus Loss of voltage		N.A.	N.A.
2) 480V Load Center undervoltage			
Load Center	Instantaneous Setpoint*	Delay Setpoint	
3A	436V(10 sec delay)	419V (60 sec ± 30 sec delay)	Within ± 5 volts of Setpoint
3B	416V(10 sec delay)	426V (60 sec ± 30 sec delay)	"
3C	417V(10 sec delay)	427V (60 sec ± 30 sec delay)	"
3D	428V(10 sec delay)	436V (60 sec ± 30 sec delay)	"
4A	415V(10 sec delay)	427V (60 sec ± 30 sec delay)	"
4B	414V(10 sec delay)	424V (60 sec ± 30 sec delay)	"
4C	401V(10 sec delay)	413V (60 sec ± 30 sec delay)	"
4D	403V(10 sec delay)	412V (60 sec ± 30 sec delay)	"
e. Trip of All Main Feedwater Pump Breakers		N.A.	N.A.

* Channel action is subject to condition being concurrent with Safety Injection signal.

TABLE 3.3-3
(Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT		TRIP SETPOINT	ALLOWABLE VALUE
7.	Loss of Power		
a.	4.16 KV Bus	N.A.	N.A.
b.	480 V Load Centers	See item 6.d.2 above for all Trip Setpoints and allowable Values	
8.	Engineered Safety Features Actuation System Interlocks		
a.	Pressurizer Pressure	$\leq$ 2000 psig	$\leq$ 2010 psig
b.	Low T_{avg}	$\geq$ 533 F	$\geq$ 531 F and $\leq$ 535 F

TABLE 4.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Safety Injection								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	R	R	1, 2, 3, 4
c. High Containment Pressure-	N.A.	R	N.A.	N.A.	M(1)	N.A.	N.A.	1, 2, 3, 4
d. Pressurizer Low Pressure-	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. High Differential Pressure Between Steam Header and Steam Lines	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
f. High Steam Line Flow - Coincident with Low Steam Line Pressure or Low T_{avg}	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
2. Containment Spray								
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	R	R	1, 2, 3, 4

TABLE 4.3-2
(Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
b. Containment Pressure--High-High Coincident with High Containment Pressure	N.A.	R	N.A.	N.A.	M(1)	N.A.	N.A.	1, 2, 3, 4
3. Containment Isolation								
a. Phase "A" Isolation								
1) Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	R	R	1, 2, 3, 4
3) Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements							
b. Phase "B" Isolation								
1) Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	R	R	1, 2, 3, 4
3) Containment Pressure--High-High Coincident with High Containment Pressure	S	R	N.A.	N.A.	M(1)	N.A.	N.A.	1, 2, 3

TABLE 4.3-2
(Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
c. Containment Ventilation Isolation								
1) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	R	R	1, 2, 3, 4
2) Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements							
3) High Containment Radioactivity	S	R	M(2)	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4
4) Manual Phase A or Phase B Containment Isolation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
4. Steam Line Isolation								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	R	R	1, 2, 3

TABLE 4.3-2
(Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
c. Containment Pressure-High-High Coincident with High Containment Pressure	S	R	N.A.	N.A.	M(1)	N.A.	N.A.	1, 2, 3, 4
d. High Steam Flow Coincident with Low Steam Line Pressure or Low T_{avg}	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
5. Feedwater Isolation								
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	R	R	R	1, 2
b. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements							
6. Auxiliary Feedwater								
a. Automatic Actuation and Actuation Relays	N.A.	N.A.	N.A.	N.A.	R	R	R	1, 2, 3

TABLE 4.3-2
(Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
b. Steam Generator Water Level--Low-Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
c. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
d. Loss of Power								
1. 4.16KV Bus Loss of Voltage (Both Busses)	N.A.	R	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
2. 480V L.C. or 4.16KV Bus undervoltage	S	R	N.A.	M(1)	N.A.	N.A.	N.A.	1, 2, 3
e. Trip of All Main Feed-water Pumps	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
7. Loss of Power								
a. 4.16 KV Bus Loss of voltage	N.A.	R	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
b. 480 V Load Center or 4.16KV Bus undervoltage	S	R	N.A.	M(1)	N.A.	N.A.	N.A.	1, 2, 3
8. Engineered Safety Features Actuation System Interlocks								
a. Pressurizer Pressure	N.A.	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
b. Low T_{avg}	N.A.	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
c. Reactor Trip	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3

TABLE 4.3-2
(Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

TABLE NOTATION

(1) Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.

(2) Analog Channel Operational Test shall consist of verification of alarm interlock and/or trip functions.

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING FOR PLANT OPERATIONS

LIMITING CONDITION FOR OPERATION

3.3.3.1 The radiation monitoring instrumentation channels for plant operations shown in Table 3.3-4 shall be OPERABLE with their Alarm/Trip Setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3-4

ACTION:

a. With a radiation monitoring channel Alarm/Trip Setpoint for plant operations exceeding the value shown in Table 3.3-4, adjust the Setpoint to within the limit within 4 hours or declare the channel inoperable.

b. With one or more radiation monitoring channels for plant operations inoperable, take the ACTION shown in Table 3.3-4.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.1 Each radiation monitoring instrumentation channel for plant operations shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST for the MODES and at the frequencies shown in Table 4.3-3.

TABLE 3.3-4

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS

FUNCTIONAL UNIT	CHANNELS TO TRIP ALARM	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	ACTION
1. Containment					
a. Containment Atmosphere Radioactivity-High (Particulate or Gaseous - See Note 1)	1	1	All	Particulate < $6.1x10^5$cpm Gaseous - (See Note 2)	A
b. RCS Leakage Detection					
1) Particulate Radioactivity	N.A.	1	1,2,3,4	N.A.	B
2) Gaseous Radioactivity	N.A.	1	1,2,3,4	N.A.	B
2. Spent Fuel Pit Areas					
Radioactivity-High Gaseous Radioactivity (Plant vent monitor for Unit 4; Spent fuel pit monitor Channel 5 for Unit 3)	1	1	*	Plant Monitor < 10^6CPM Spent fuel pit monitor < $5.5x10^2$ $\frac{\mu ci}{cc}$	D
3. Control Room ** Air Intake Radiation Level	1	1	All	< 2mR/hr	C

TABLE 3.3-4 (Continued)

TABLE NOTATIONS

** This item does not apply until after implementation of PC/M 83-170 and PC/M 84-26

* With irradiated fuel in the spent fuel pits.

Note 1 Either the particulate or gaseous channel in the OPERABLE status will satisfy this LCO.

Note 2 Containment Gaseous Monitor Setpoint= $\dfrac{(3.2 \times 10^4)}{(F)}$ CPM,

Where F = $\dfrac{\text{Actual Purge Flow}}{\text{Design Purge Flow (35,000 CFM)}}$

Setpoint may vary according to current plant conditions provided that the release rate does not exceed allowable limits provided in Specification 3.11.2.1.

ACTION STATEMENTS

ACTION A - With the number of OPERABLE channels less than the Minimum Channels OPERABLE requirement, operation may continue provided containment purge is terminated and the containment purge and exhaust valves and the instrument air bleed valves are maintained closed.

ACTION B - Must satisfy the ACTION requirement for Specification 3.4.6.1.

ACTION C - With the number of OPERABLE channels less than the Minimum Channels OPERABLE requirement, within 1 hour isolate the Control Room Ventilation System from its outside air supply and initiate operation of the Control Room Ventilation System in the recirculation mode.

ACTION D - With the number of OPERABLE channels less than the Minimum Channels OPERABLE requirement, immediately suspend operations in the Spent Fuel Pit area involving spent fuel manipulation.

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Containment				
a. Containment Atmosphere Radioactivity-High	S	R	M	All
b. RCS Leakage Detection				
1) Particulate Radioactivity	S	R	M***	1, 2, 3, 4
2) Gaseous Radioactivity	S	R	M***	1, 2, 3, 4
2. Spent Fuel Pit Areas				
Radioactivity-High Gaseous Radioactivity	S	R	M	*
3. Control Room ** Air Intake Radiation Level	S	R	M	All

TABLE NOTATIONS

* With irradiated fuel in the spent fuel storage pits.

** This item does not apply until after implementation of PC/M 83-170 and PC/M 84-26

*** Analog Channel Operational Test shall consist of verification of alarm interlock and/or trip functions

MOVABLE INCORE DETECTORS

LIMITING CONDITION FOR OPERATION

3.3.3.2 The movable Incore Detection System shall be OPERABLE with:

a. At least 75% of the detector thimbles,

b. A minimum of two detector thimbles per core quadrant, and

c. Sufficient movable detectors, drive, and readout equipment to map these thimbles.

APPLICABILITY: When the Movable Incore Detection System is used for:

a. Recalibration of the Excore Neutron Flux Detection System, or

b. Monitoring the QUADRANT POWER TILT RATIO, or

c. Measurement of $F^N_{\Delta H}$ and $F_Q(Z)$

ACTION:

With the Movable Incore Detection System inoperable, do not use the system for the above applicable monitoring or calibration functions. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.2 The Movable Incore Detection System shall be demonstrated OPERABLE at least once per 24 hours by normalizing each detector output when required for:

a. Recalibration of the Excore Neutron Flux Detection System, or

b. Monitoring the QUADRANT POWER TILT RATIO, or

c. Measurement of $F^N_{\Delta H}$ and $F_Q(Z)$

INSTRUMENTATION

SEISMIC INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.3 The seismic monitoring instrumentation (SEISMOGRAPH) shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With the above required seismic monitoring instrument inoperable for more than 30 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the instrument to OPERABLE status.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.3.1 The above seismic monitoring instrument shall be demonstrated OPERABLE:

a. At least once per 92 days by performance of a battery test and trace check on the seismograph

b. At least once per 12 months by replacing film for the seismic monitoring instrument with fresh film.

c. At least once per 18 months by performance of a CHANNEL CALIBRATION.

4.3.3.3.2 If the above required seismic monitoring instrument is actuated during a seismic event greater than or equal to 0.01g, it shall be restored to OPERABLE status within 24 hours and a CHANNEL CALIBRATION performed within 10 days following the seismic event. Data shall be retrieved from actuated instruments and analyzed to determine the magnitude of the vibratory ground motion. A special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 14 days describing the magnitude, frequency spectrum, and resultant effect upon facility features important to safety.

INSTRUMENTATION

METEOROLOGICAL INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.4 The meteorological monitoring instrumentation channels shown in Table 3.3-5 shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more required meteorological monitoring channels inoperable for more than 7 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the channel(s) to OPERABLE status.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.4 Each of the above meteorological monitoring instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK and CHANNEL CALIBRATION at the frequencies shown in Table 4.3-4.

TABLE 3.3-5

METEOROLOGICAL MONITORING INSTRUMENTATION

INSTRUMENT	LOCATION	MINIMUM OPERABLE
1. Wind Speed		
a. 10 Meter Wind Speed	Nominal Elev. 30 feet	1*
b. 60 Meter Wind Speed	Nominal Elev. 195 feet	N.A.
2. Wind Direction		
a. 10 Meter Wind Direction	Nominal Elev. 30 feet	1*
b. 60 Meter Wind Direction	Nominal Elev. 195 feet	N.A.
3. Air Temperature - ΔT		
a. Delta T 'A'	Sensor locations: 32 ft., 195 ft.	1**
b. Delta T 'B'	Sensor Locations: 32 ft., 195 ft.	N.A.

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* The 60 meter channel may be substituted for the 10-meter wind speed and/or direction channel for up to 30 days in the event the 10-meter channel is inoperable. Wind speed data from the 60 meter elevation should be adjusted using the wind speed power law:

$$S_{10 \text{ meters}} = S_{60 \text{ meters}} (0.1667)^n$$

where:

S = wind speed in mph
n = 0.25 for Pasquill Vertical Stability Classes A,B,C,D.
n = 0.50 for Pasquill Vertical Stability Classes E,F,G.
1.667×10^{-1} = constant = 10 meters/60 meters

** OPERABILITY OF ONE OF THE TWO DELTA T CHANNELS MEET MINIMUM OPERABILITY REQUIREMENT

TABLE 4.3-4

METEOROLOGICAL MONITORING INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Wind Speed		
a. Nominal Elev. 30 feet	D	SA
b. Nominal Elev. 195 feet	D*	SA*
2. Wind Direction		
a. Nominal Elev. 30 feet	D	SA
b. Nominal Elev. 195 feet	D*	SA*
3. Air Temperature - ΔT*		
a. Delta T 'A'	D	SA
b. Delta T 'B'	D*	SA*

*CHANNEL CHECK AND CHANNEL CALIBRATION SHALL BE PERFORMED ON AT LEAST ONE OPERABLE CHANNEL PER MINIMUM OPERABILITY REQUIREMENTS OF TABLE 3.3-5.

INSTRUMENTATION

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.5 The accident monitoring instrumentation channels shown in Table 3.3-6 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3-6.

ACTION:

a. With the number of OPERABLE accident monitoring instrumentation channels less than the Total Number of Channels or the MINIMUM CHANNELS OPERABLE shown in Table 3.3-6, comply with appropriate action requirements shown in Table 3.3-6.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.5 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, ANALOG CHANNEL OPERATIONAL TEST, and CHANNEL CALIBRATION at the frequencies shown in Table 4.3-5.

TABLE 3.3-6

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENTATION	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	APPLICABLE ACTIONS
1. Pressurizer Water Level	2	1	1,2,3	1,2
2. Reactor Coolant Pressure (Wide Range)	2	1	1,2,3	1,2
3. Reactor Coolant Outlet Temperature - T_{HOT} (Wide Range)	2/2 detectors per channel	1/2 detectors per channel	1,2,3	1,2
4. Reactor Coolant Inlet Temperature - T_{COLD} (Wide Range)	2/2 detectors per channel	1/2 detectors per channel	1,2,3	1,2
5. Auxiliary Feedwater Flow Rate	2/ generator	1/ generator	1,2,3	1,2
6. Reactor Coolant System Subcooling Margin Monitor	2(2)	1(2)	1,2,3	1,2
7. PORV Position Indicator (Primary Detector)	1/ valve	1/ valve	1,2,3	3
8. PORV Block Valve Position	1/ valve	1/ valve	1,2,3	3
9. Safety Valve Position Indicator (Primary Detector)	1/ valve	1/ valve	1,2,3	1,2
10. Containment Pressure (Wide Range)	2	1	1,2,3	1,2
11. Containment Pressure (Narrow Range)	2	1	1,2,3	6
12. Containment Water Level (Wide Range)	2	1	1,2,3	1,2
13. Containment Water Level (Narrow Range)	2	1	1,2,3	6

TABLE 3.3-6 (Continued)

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENTATION	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	APPLICABLE ACTIONS
14. Containment High Range Area Radiation	2	1	1,2,3	4
15. Containment Hydrogen Monitors	2	1	See Spec. 3.6.5	5
16. High Range - Noble Gas Effluent Monitors				
a. Plant Vent Exhaust	1	1	ALL	4
b. Unit 3 - Spent Fuel Pit Exhaust	1	1	ALL	4
c. Condenser Air Ejectors	1	1	1,2,3	4
d. Main Steam Lines	1	1	1,2,3	4
17. Incore Thermocouples (Core Exit Thermocouples)	4/core quadrant (2 Per indep. channel)	2/core quadrant (1 per indep. channel)	1,2,3	1,2
18. Refueling Water Storage Tank Water Level	2	1	1,2,3	1,2
19. Boric Acid Tank Solution Level	1(3)	1	1,2,3	1,2
20. Reactor Vessel Level Monitoring System	2(1)	1(1)	1,2,3	7,8

TABLE NOTATIONS

(1) A channel is eight sensors in a probe. A channel is operable if a minimum of four sensors are operable.

(2) Inputs per items 2, 3, 4 and 17.

(3) Applicable for inservice Boric Acid Tank(s) only.

TABLE 3.3-6 (Continued)

ACTION STATEMENTS

ACTION 1 With the number of OPERABLE accident monitoring instrumentation channel(s) less than the Total Number of Channels either restore the inoperable channel(s) to OPERABLE status within 7 days, or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

ACTION 2 With the number of OPERABLE accident monitoring instrumentation channels less than the minimum channels OPERABLE, either restore the inoperable channel(s) to OPERABLE status within 48 hours, or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

ACTION 3 Close the associated block valve and open its circuit breaker.

ACTION 4 With the number of OPERABLE Channels less than required by the Minimum Channels OPERABLE requirements, initiate the preplanned alternate method of monitoring the appropriate parameter(s), within 72 hours, and:

1) either restore the inoperable channel(s) to OPERABLE status within 7 days of the event, or

2) prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the system to OPERABLE status.

ACTION 5 With one or both hydrogen monitor(s) inoperable, comply with Action requirements of Specification 3.6.5.

ACTION 6 With the number of OPERABLE accident monitoring instrumentation channels less than the minimum channel OPERABLE, either restore the inoperable channel to OPERABLE status within 30 days, or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

ACTION 7 With the number of OPERABLE channels one less than the total number of channels, restore the system to OPERABLE status within 7 days. If repairs are not feasible without shutting down, prepare and submit a Special Report to the commission pursuant to Specification 6.9.2 within 30 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

TABLE 3.3-6 (Continued)

ACCIDENT MONITORING INSTRUMENTATION

ACTION 8 With the number of OPERABLE channels less than the Minimum Channels OPERABLE requirements, restore the inoperable channel(s) to OPERABLE status within 48 hours. If repairs are not feasible without shutting down:

1. Initiate an alternate method of monitoring the reactor vessel inventory; and

2. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status; and

3. Restore at least one channel to OPERABLE status at the next scheduled refueling.

TABLE 4.3-5

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

	INSTRUMENTATION	CHANNEL CHECK	ANALOG CHANNEL OPERATIONAL TEST	CHANNEL CALIBRATION
1.	Pressurizer Water Level	S	M	R
2.	Reactor Coolant Pressure (Wide Range)	M	NA	R
3.	Reactor Coolant Outlet Temperature T_{HOT} (Wide Range)	M	NA	R
4.	Reactor Coolant Inlet Temperature T_{COLD} (Wide Range)	M	NA	R
5.	Auxiliary Feedwater Flow Rate	M	NA	R
6.	Reactor Coolant System Subcooling Margin Monitor	M	NA	R
7.	PORV Position Indicator (Primary Detector)	M	R(1)	NA
8.	PORV Block Valve Position Detector	M	R(1)	NA
9.	Safety Valve Position Indicator (Primary Detector)	M(4)	NA	R
10.	Containment Pressure (Wide Range)	M(3)	NA	R
11.	Containment Pressure (Narrow Range)	M(3)	NA	R
12.	Containment Water Level (Wide Range)	M(3)	NA	R
13.	Containment Water Level (Narrow Range)	M(3)	NA	R
14.	Containment High Range Area Radiation	S(3)	NA	R(2)
15.	Containment Hydrogen Monitors	[as required by Specification 4.6.5.1]		

TABLE 4.3-5 (Continued)

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

	INSTRUMENTATION	CHANNEL CHECK	ANALOG CHANNEL OPERATIONAL TEST	CHANNEL CALIBRATION
16.	High Range - Noble Gas Effluent monitors			
	a. Plant Vent Exhaust	S(5)	M	R
	b. Unit 3 - Spent Fuel Pit Exhaust	S(5)	M	R
	c. Condenser Air Ejectors	S(5)	M	R
	d. Main Steam Lines	S(5)	M	R
17.	Incore Thermocouples (Core Exit Thermocouples)	M(3)	NA	R
18.	Refueling Water Storage Tank Water Level	M(3)	R	R
19.	Boric Acid Tank Solution Level	M(3)	R	R
20.	Reactor Vessel Level Monitoring System	M(3)	NA	R

(1) These are not analog channels. The requirement is satisfied if the channel test includes the verification and necessary adjustment of the channel such that the channel output results in the proper indications and/or alarm functions.

(2) Acceptable criteria for calibration is provided in Table II.F.1-3 of NUREG-0737.

(3) In addition to requirements in Modes 1, 2, and 3, the surveillance for these channels shall be performed within one surveillance interval prior to heatup above 200 F.

(4) By observation of TEC monitor power light ON' and test of the associated alarm.

(5) By observation of related parameters.

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.6 The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.3-7 shall be OPERABLE with their Alarm/Trip Setpoints set to ensure that the limits of Specifications 3.11.1.1 are not exceeded. The Alarm/Trip Setpoints of these channels shall be determined and adjusted in accordance with the methodology and parameters in the ODCM.

APPLICABILITY: At all times

ACTION.

a. With a radioactive liquid effluent monitoring instrumentation channel Alarm/Trip Setpoint less conservative than required by the above specification, immediately suspend the release of radioactive effluents monitored by the affected channel, or declare the channel inoperable, or change the setpoint so it is acceptably conservative,

b. With less than the minimum number of radioactive liquid effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3-7. Restore the inoperable instrumentation to OPERABLE status within the time specified in the ACTION, or explain in the next Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.4 why this inoperability was not corrected within the time specified.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.6 Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST at the frequencies shown in Table 4.3-6.

TABLE 3.3-7

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
1. Gross Radioactivity Monitors Providing Alarm and Automatic Termination of Release		
a. Liquid Radwaste Effluent Line	1	A
b. Steam Generator Blowdown Effluent Line	1	B
2. Flow Rate Measurement Devices		
a. Liquid Radwaste Effluent Line	1	C
b. Steam Generator Blowdown Effluent Line	1	C

TABLE 3.3-7 (Continued)

ACTION STATEMENTS

ACTION A - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 14 days provided that prior to initiating a release:

a. At least two independent samples are analyzed in accordance with Specification 4.11.1.1.1, and

b. At least two technically qualified members of the facility staff independently verify the release rate calculations and discharge line valving.

Otherwise, suspend release of radioactive effluents via this path-way.

ACTION B - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided grab samples are analyzed for gross beta/gamma radioactivity at a lower limit of detection of no more than 10^{-7} microCurie/ml; or analyzed isotopically at a limit of detection of at least 5×10^{-7} microCurie/ml:

a. At least once per 12 hours when the specific activity of the secondary coolant is greater than 0.01 microCurie/gram DOSE EQUIVALENT I-131, or

b. At least once per 24 hours when the specific activity of the secondary coolant is less than or equal to 0.01 microCurie/gram DOSE EQUIVALENT I-131.

ACTION C - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided the flow rate is estimated at least once per 4 hours during actual releases. Pump performance curves may be used to estimate flow.

TABLE 4.3-6

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL* CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST
1. Radioactivity Monitors Providing Alarm and Automatic Termination of Release			
a. Liquid Radwaste Effluent Line	D,P	R(2)	Q(1)
b. Steam Generator Blowdown Effluent Line	D	R(2)	Q(1)
2. Flow Rate Measurement Devices			
a. Liquid Radwaste Effluent Line	D(3)	R	Q
b. Steam Generator Blowdown Effluent Line	D(3)	R	Q

*Channel calibration frequency shall be at least once per 18 months.

TABLE 4.3-6 (Continued)

TABLE NOTATIONS

(1) The ANALOG CHANNEL OPERATIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occur if the following condition exists:

a. Instrument indicates measured levels above the Alarm/Trip Setpoint

(2) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards (NBS) or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used.

(3) CHANNEL CHECK shall consist of verifying indication of flow during periods of release. CHANNEL CHECK shall be made at least once per 24 hours on days on which continuous, periodic, or batch releases are made.

INSTRUMENTATION

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.7 The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.3-8 shall be OPERABLE with their Alarm/Trip Setpoints set to ensure that the limits of Specifications 3.11.2.1 and 3.11.2.5 are not exceeded. The Alarm/Trip Setpoints of these channels meeting Specification 3.11.2.1 shall be determined and adjusted in accordance with the methodology and parameters in the ODCM.

APPLICABILITY: As shown in Table 3.3-8

ACTION

a. With a radioactive gaseous effluent monitoring instrumentation channel Alarm/Trip Setpoint less conservative than required by the above specification, immediately suspend the release of radioactive gaseous effluents monitored by the affected channel, or declare the channel inoperable, or change the setpoint so it is acceptably conservative,

b. With the number of OPERABLE radioactive gaseous effluent monitoring instrumentation channels less than the Minimum Channels OPERABLE, take the ACTION shown in Table 3.3-8. Restore the inoperable instrumentation to OPERABLE status within the time specified in the. ACTION, or explain in the next Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.4 why this inoperability was not corrected within the time specified.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.7 Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST at the frequencies shown in Table 4.3-7..

TABLE 3.3-8

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
1. GAS DECAY TANK SYSTEM			
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release (Plant Vent Monitor)	1	*	45
2. GAS DECAY TANK SYSTEM Explosive Gas Monitoring System			
a. Hydrogen and Oxygen Monitors	1	**	49
3. Condenser Air Ejector Vent System			
a. Noble Gas Activity Monitor	1	***	47
b. Iodine Sampler	1	***	48
c. Particulate Sampler	1	***	48
d. Flow Rate Monitor	1	***	46
e. Sampler Flow Rate Monitor	1	***	46
4. Plant Vent System			
a. Noble Gas Activity Monitor	1	*	47
b. Iodine Sampler	1	*	48
c. Particulate Sampler	1	*	48
d. Sampler Flow Rate Monitor	1	*	46
5. Unit 3 Spent Fuel Pit Building Vent			
a. Noble Gas Activity Monitor	1	****	47
b. Iodine Sampler	1	****	48
c. Particulate Sampler	1	****	48
d. Sampler Flow Rate Monitor	1	****	46

TABLE 3.3-8 (Continued)

TABLE NOTATIONS

* At all times.

** During GAS DECAY TANK SYSTEM operation.

*** Applies during Modes 1, 2, 3, and 4.

**** Applies at all times when irradiated fuel is in the Unit 3 spent fuel pit.

ACTION STATEMENTS

ACTION 45 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, the contents of the tank(s) may be released to the environment for up to 14 days provided that prior to initiating the release:

a. At least two independent samples of the tank's contents are analyzed, and

b. At least two technically qualified members of the facility staff independently verify the release rate calculations and discharge valve lineup.

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 46 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided the flow rate is estimated at least once per 4 hours.

ACTION 47 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided grab samples are taken at least once per 12 hours and these samples are analyzed for radioactivity within 24 hours.

ACTION 48 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via the affected pathway may continue for up to 30 days provided samples are continuously collected with auxiliary sampling equipment as required in Table 4.11-2 and analyzed at least 4 times a month.

TABLE 3.3-8 (Continued)

TABLE NOTATIONS (Continued)

ACTION 49 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, operation of the GAS DECAY TANK SYSTEM may continue for up to 30 days provided that grab samples are collected and analyzed for hydrogen and oxygen concentration at least a) once per 4 hours during degassing operations, and b) once per 24 hours during other operations.

Until modifications to the gas analyzing equipment are complete, the following action will be taken:

With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, immediately initiate corrective actions to restore the channel to OPERABLE status.

TABLE 4.3-7

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. GAS DECAY TANK SYSTEM				
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Rèlease (Plant Vent Monitor)	P	R(3)	Q(1)	*
2. GAS DECAY TANK SYSTEM Explosive Gas Monitoring System				
a. Hydrogen and Oxygen Monitors	D	Q(4) and Q(5)	M	**
3. Condenser Air Ejector Vent				
a. Noble Gas Activity Monitor	D	R(3)	Q(2)	***
b. Iodine Sampler	W	N/A	N/A	****
c. Particulate Sampler	W	N/A	N/A	****
d. Flow Rate Monitor	D	R	N/A	****
e. Sampler Flow Rate Monitor	D	R	N/A	****
4. Plant Vent System				
a. Noble Gas Activity Monitor	D	(3),(6)	Q(2)	*
b. Iodine Sampler	W	N/A	N/A	*
c. Particulate Sampler	W	N/A	N/A	*
d. Sampler Flow Rate Monitor	D	R	N/A	*

TABLE 4.3-7 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
5. Unit 3 Spent Fuel Pit Building Vent System				
a. Noble Gas Activity Monitor	D	R(3)	Q	*****
b. Iodine Sampler	W	N/A	N/A	*****
c. Particulate Sampler	W	N/A	N/A	*****
d. Sampler Flow Rate Monitor	D	R	N/A	*****

TABLE 4.3-7 (Continued)

TABLE NOTATIONS

* At all times.

** During GAS DECAY TANK SYSTEM operation.

*** Applies during Modes 1, 2, 3, and 4.

**** Applies during Modes 1,2,3, and 4 when primary to secondary leakage is detected.

***** Applies at all times when irradiated fuel is in Unit 3 spent fuel pit.

(1) The ANALOG CHANNEL OPERATIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occurs if the following conditions exist:

a. Instrument indicates measured levels above the Alarm/Trip Setpoint.

(2) The ANALOG CHANNEL OPERATIONAL TEST shall also demonstrate that control room alarm annunciation occurs if the following condition exists:

a. Instrument indicates measured levels above the Alarm Setpoint.

(3) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards (NBS) or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used.

(4) The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal:

a. One volume percent hydrogen, balance nitrogen, and

b. Four volume percent hydrogen, balance nitrogen.

(5) The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal:

a. One volume percent oxygen, balance nitrogen, and

b. Four volume percent oxygen, balance nitrogen.

(6) The CHANNEL CALIBRATION frequency shall be at least once per 18 months.

3/4.3.4 TURBINE OVERSPEED PROTECTION

LIMITING CONDITION FOR OPERATION

3.3.4 At least one Turbine Overspeed Protection System shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one stop valve or one governor valve per high pressure turbine steam line inoperable and/or with one reheat stop valve or one reheat intercept valve per low pressure turbine steam line inoperable, restore the inoperable valve(s) to OPERABLE status within 72 hours, or close at least one valve in the affected steam line(s) or isolate the turbine from the steam supply within the next 6 hours.

b. The provisions of Specification 3.0.4 do not apply for entry into Modes 2 and 3 if all MSIVs are closed.

SURVEILLANCE REQUIREMENTS

4.3.4.1 The provisions of Specification 4.0.4 are not applicable.

4.3.4.2 The above required Turbine Overspeed Protection System shall be demonstrated OPERABLE:

a. At least once per 31 days by cycling each of the following valves through at least one complete cycle from the running position:

   1) Two high pressure turbine stop valves,

   2) Four high pressure turbine governor valves,

   3) Four low pressure turbine reheat stop valves, and

   4) Four low pressure turbine reheat intercept valves.

b. At least once per 31 days by direct observation of the movement of each of the above valves through one complete cycle from the running position,

c. At least once per refueling by performance of a CHANNEL CALIBRATION on the Turbine Overspeed Protection Systems, and

SURVEILLANCE REQUIREMENTS (Continued)

d. At least once per 40 months by disassembling at least one of each of the above valves and performing a visual and surface inspection of valve seats, disks, and stems and verifying no unacceptable flaws or excessive corrosion. If unacceptable flaws or excessive corrosion are found, all other valves of that type shall be inspected.

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

STARTUP AND POWER OPERATION

LIMITING CONDITION FOR OPERATION

3.4.1.1 All reactor coolant loops shall be in operation.

APPLICABILITY: MODES 1 and 2.*

ACTION:

With less than the above required reactor coolant loops in operation, be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.1.1 The above required reactor coolant loops shall be verified in operation and circulating reactor coolant at least once per 12 hours.

*See Special Test Exceptions Specification 3.10.4

REACTOR COOLANT SYSTEM

HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.1.2 At least two of the reactor coolant loops listed below shall be OPERABLE with two reactor coolant loops in operation when the Reactor Trip System breakers are closed and one reactor coolant loop in operation when the Reactor Trip System breakers are open:*

a. Reactor Coolant Loop A and its associated steam generator and reactor coolant pump,

b. Reactor Coolant Loop B and its associated steam generator and reactor coolant pump,

c. Reactor Coolant Loop C and its associated steam generator and reactor coolant pump.

APPLICABILITY: MODE 3

ACTION:

a. With less than the above required reactor coolant loops OPERABLE, restore the required loops to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

b. With only one reactor coolant loop in operation and the Reactor Trip system breakers in the closed position, within 1 hour open the Reactor Trip System breakers.

c. With no reactor coolant loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required reactor coolant loop to operation.

---

* All reactor coolant pumps may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10 F below saturation temperature.

REACTOR COOLANT SYSTEM

HOT STANDBY

SURVEILLANCE REQUIREMENTS

4.4.1.2.1 At least the above required reactor coolant pumps, if not in operation, shall be determined OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.2.2 The required steam generators shall be determined OPERABLE by verifying secondary side water level to be greater than or equal to 10% at least once per 12 hours.

4.4.1.2.3 The required reactor coolant loops shall be verified in operation and circulating reactor coolant at least once per 12 hours.

REACTOR COOLANT SYSTEM

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.3 At least two of the loops listed below shall be OPERABLE and at least one of these loops shall be in operation:*

a. Reactor Coolant Loop A and its associated steam generator and reactor coolant pump,**

b. Reactor Coolant Loop B and its associated steam generator and reactor coolant pump,**

c. Reactor Coolant Loop C and its associated steam generator and reactor coolant pump,**

d. RHR loop A, and***

e. RHR loop B.***

APPLICABILITY: MODE 4.

ACTION:

a. With less than the above required loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible; if the remaining OPERABLE loop is an RHR loop, be in COLD SHUTDOWN within 24 hours.

b. With no loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required loop to operation.

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* All reactor coolant pumps and RHR pumps may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10 F below saturation temperature.

** A reactor coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 275 F unless the secondary water temperature of each steam generator is less than 50 F above each of the Reactor Coolant System cold leg temperatures (including instrument error).

*** An inoperable RHR loop may be the operating RHR loop for up to 2 hours for surveillance testing to establish operability.

SURVEILLANCE REQUIREMENTS

4.4.1.3.1 The required coolant loop pump(s), if not in operation, shall be determined OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.3.2 The required steam generator(s) shall be determined OPERABLE by verifying secondary side water level to be greater than or equal to 10% at least once per 12 hours.

4.4.1.3.3 At least one reactor coolant or RHR loop shall be verified in operation and circulating reactor coolant at least once per 12 hours.

REACTOR COOLANT SYSTEM

COLD SHUTDOWN - LOOPS FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.1 At least one residual heat removal (RHR) loop shall be OPERABLE and in operation,* and

a. One additional RHR loop shall be OPERABLE,** or

b. The secondary side water level of at least two steam generators shall be greater than 10%.

APPLICABILITY: MODE 5 with reactor coolant loops filled.***

ACTION:

a. With one of the RHR loops inoperable and with less than the required steam generator water level, immediately initiate corrective action to return the inoperable RHR loop to OPERABLE status or restore the required steam generator water level as soon as possible.

b. With no RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to operation.

* The RHR pump may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10 F below saturation temperature.

** An inoperable RHR loop may be the operating RHR loop for up to 2 hours for surveillance testing to establish operability.

*** A reactor coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 275 F unless the secondary water temperature of each steam generator is less than 50 F above each of the Reactor Coolant System cold leg temperatures.

SURVEILLANCE REQUIREMENTS

4.4.1.4.1.1 The secondary side water level of at least two steam generators when required shall be determined to be within limits at least once per 12 hours.

4.4.1.4.1.2 At least one RHR loop shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

4.4.1.4.1.3 The RHR pump not operating, when required, shall be determined OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

REACTOR COOLANT SYSTEM

COLD SHUTDOWN - LOOPS NOT FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.2 Two residual heat removal (RHR) loops shall be OPERABLE* and at least one RHR loop shall be in operation.**

APPLICABILITY: MODE 5 with reactor coolant loops not filled.

ACTION:

a. With less than the above required RHR loops OPERABLE, immediately initiate corrective action to return the required RHR loops to OPERABLE status as soon as possible.

b. With no RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.4.2.1 At least one RHR loop shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

4.4.1.4.2.2 The RHR pump not operating shall be determined OPERABLE once per 7 days by verifying correct breaker alignments and indicated availability.

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* An inoperable RHR loop may be the operating RHR loop for up to 2 hours for surveillance testing to establish operability.

** The RHR pump may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10 F below saturation temperature.

REACTOR COOLANT SYSTEM

3/4.4.2 SAFETY VALVES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2.1 As a minimum one of the following shall be OPERABLE to prevent the RCS from being pressurized above its safety limit:

a. One pressurizer Code safety with a lift setting of 2485 psig ± 1%*, or

b. Applicable in Mode 5 only, one Code safety valve removed to create a minimum opening of 2.20 square inches.

APPLICABILITY: MODES 4 and 5.

ACTION:

With no pressurizer Code Safety valve OPERABLE, immediately suspend all operations involving positive reactivity changes and depressurize and vent the RCS through an opening with an area of at least 2.20 square inches.

SURVEILLANCE REQUIREMENTS

4.4.2.1 No additional requirements other than those required by Specification 4.0.5.

* The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

REACTOR COOLANT SYSTEM

OPERATING

LIMITING CONDITION FOR OPERATION

3.4.2.2 All pressurizer Code safety valves shall be OPERABLE with a lift setting of 2485 psig ± 1%*

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With one pressurizer Code Safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.2.2. Test each valve to the requirements of Specification 4.0.5.

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* The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

REACTOR COOLANT SYSTEM

3/4.4.3 PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.3 The pressurizer shall be OPERABLE with a steam bubble with a water volume of less than or equal to 92% of indicated level, and at least two group of pressurizer heaters having a capacity of at least 125KW capable of being supplied by emergency power.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With only one group of pressurizer heaters OPERABLE, restore at least two groups to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With the pressurizer otherwise inoperable, be in at least HOT STANDBY with the Reactor Trip System breakers open within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.1 The pressurizer water volume shall be determined to be within its limit at least once per 12 hours.

4.4.3.2 The capacity of the above required groups of pressurizer heaters shall be verified by energizing the heaters and measuring circuit current at least once per 92 days.

4.4.3.3 The emergency power supply for the pressurizer heaters shall be demonstrated OPERABLE at least once per refueling by demonstrating the capability to power the heaters from the emergency power.

REACTOR COOLANT SYSTEM

3/4.4.4 PORV BLOCK VALVES

LIMITING CONDITION FOR OPERATION

3.4.4 Each Power Operated Relief Valve (PORV) Block valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one or more block valve(s) inoperable, within 1 hour either restore the block valve(s) to OPERABLE status or close the block valve(s) and remove power from the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.4 Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel unless the block valve is closed with power removed in order to meet the requirements of Action a. in Specification 3.4.4.

3/4.4.5 STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

3.4.5 Each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more steam generators inoperable, restore the inoperable generator(s) to OPERABLE status prior to increasing T_{avg} above 200 F.

SURVEILLANCE REQUIREMENTS

4.4.5.0 Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

4.4.5.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the one steam generator. The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 3 N % of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

4.4.5.2 Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-1. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.5.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.5.4. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas;

b. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

   1) All nonplugged tubes that previously had detectable wall penetrations (greater than 20%),

   2) Tubes in those areas where experience has indicated potential problems, and

   3) A tube inspection (pursuant to Specification 4.4.5.4a.8) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

c. The tubes selected as the second and third samples (if required by Table 4.4-1) during each inservice inspection may be subjected to a partial tube inspection provided:

   1) At least 50% of the tubes selected for these samples are the tubes from those areas of the tube sheet array where tubes with imperfections were previously found, and

   2) The inspections include those portions of the tubes where imperfections were previously found.

The results of each sample inspection shall be classified into one of the following three categories:

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.

C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

Note: In all inspections, previously degraded tubes must exhibit significant (greater than 10%) further wall penetrations to be included in the above percentage calculations.

4.4.5.3 Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months following replacement of steam generators. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections following service under AVT conditions, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months;

b. If the results of the inservice inspection of a steam generator conducted in accordance with Table 4.4-1 at 40-month intervals fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 4.4.5.3a; the interval may then be extended to a maximum of once per 40 months; and

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.4-1 during the shutdown subsequent to any of the following conditions:

   1) primary-to-secondary tubes leak (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.6.2, or

2) A seismic occurrence greater than the Operating Basis Earthquake, or

3) A loss-of-coolant accident requiring actuation of the Engineered Safety Features, or

4) A main steam line or feedwater line break requiring actuation of the automatic steam and feedwater isolation valves.

4.4.5.4 Acceptance Criteria

a. As used in this specification:

1) Imperfection means an exception to the dimensions, finish, or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections;

2) Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube;

3) Degraded Tube means a tube containing imperfections greater than or equal to 20% of the nominal wall thickness caused by degradation;

4) % Degradation means the percentage of the tube wall thickness affected or removed by degradation;

5) Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective. Any tube which does not permit the passage of the eddy current inspection probe shall be deemed a defective tube.

6) Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service and is equal to 40% of the nominal tube wall thickness;

7) Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in Specification 4.4.5.3c above;

8) Tube Inspection means an inspection of the steam generator tube from the point of entry completely around the U-bend to the top support of the opposite leg; and

9) Preservice Inspection means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service following steam generator replacement to establish a baseline condition of the tubing.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 4.4-1.

4.4.5.5 Reports

a. Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2;

b. The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a Special Report pursuant to Specification 6.9.2 within 12 months following the completion of the inspection. This Special Report shall include:

1) Number and extent of tubes inspected,

2) Location and percent of wall-thickness penetration for each indication of an imperfection, and

3) Identification of tubes plugged.

c. In addition to notification requirements per 10CFR50.72 as specified in Table 4.4-1, results of steam generator tube inspections which fall into Category C-3 shall be reported in a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days and prior to resumption of plant operation. This report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

TABLE 4.4-1
STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S.G.	C-1	None	N.A.	N.A.	N.A.	N.A.
	C-2	Plug defective tubes and inspect additional 2S tubes in this S.G.	C-1	None	N.A.	N.A.
			C-2	Plug defective tubes and inspect additional 4S tubes in this S.G.	C-1	None
					C-2	Plug defective tubes
					C-3	perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first Sample	N.A.	N.A.
	C-3	Inspect all tubes in this S.G., plug defective tubes and inspect 25 tubes in each other S.G. Notification to NRC pursuant to 50.72 (b)(2) of 10 CFR Part 50	All other S.G.s are C-1	None	N.A.	N.A.
			Some S.G.s C-2 but no additional S.G. are C-3	Perform action for C-2 result of second sample	N.A.	N.A.
			Additional S.G. is C-3	Inspect all tubes in each S.G. and plug defective tubes. Notification to NRC pursuant to 50.72 (b)(2) of 10 CFR Part 50	N.A.	N.A.

$S = 3\,\frac{N}{n}\,\%$ Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection.

REACTOR COOLANT SYSTEM

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.6.1 The following Reactor Coolant System Leakage Detection Systems shall be OPERABLE:

a. The Containment Atmosphere Particulate Radioactivity Monitoring System,

b. A Containment Sump Level Monitoring System, and

c. The Containment Atmosphere Gaseous Radioactivity Monitoring System.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With either the Particulate or Gaseous Radioactive Monitoring System not OPERABLE, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed at least once per 24 hours, otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. (See Specifications 3/4.9.13 and 3/4.3.3)

b. With both the above required radioactive monitoring leakage detection systems inoperable, operations may continue for up to 7 days provided:

   1. A reactor cavity sump level monitoring system is OPERABLE,

   2. Appropriate grab samples are obtained and analyzed at least once per 24 hours, and

   3. A Reactor Coolant System water inventory balance is performed at least once per 8 hours during steady state operation except when operating in the shutdown cooling mode;

   otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With no Containment Sump Level Monitoring System operable, restore at least one system to OPERABLE status within 7 days, or be in at least HOT STANDBY within the next 6 hours and in at least Hot Shutdown within the following 6 hours.

REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

SURVEILLANCE REQUIREMENTS

4.4.6.1 The Leakage Detection Systems shall be demonstrated OPERABLE by:

a. Containment Atmosphere Gaseous and Particulate Monitoring Systems - performance of CHANNEL CHECK, CHANNEL CALIBRATION, and ANALOG CHANNEL OPERATIONAL TEST at the frequencies specified in Table 4.3-3,

b. Containment Sump Level Monitoring System-performance of CHANNEL CHECK at least once per 31 days and CHANNEL CALIBRATION at least once per refueling.

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 gpm UNIDENTIFIED LEAKAGE,

c. 1 gpm total reactor-to-secondary leakage through all steam generators not isolated from the Reactor Coolant System and 500 gallons per day through any one steam generator,

d. 10 gpm IDENTIFIED LEAKAGE from the Reactor Coolant System, and

e. Leakage as specified in Table 3.4-1 at a Reactor Coolant system pressure of 2235 ± 20 psig from any Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1.*

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE and leakage from Reactor Coolant System Pressure Isolation Valves, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

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* Test pressures less than 2235 psig are allowed. Minimum differential test pressure shall not be less than 150 psid. Observed leakage shall be adjusted for the actual test pressure up to 2235 psig assuming the leakage to be directly proportional to pressure differential to the one-half power.

LIMITING CONDITION FOR OPERATION (CONTINUED)

ACTION (continued):

c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than allowed by 3.4.6.2.e above operation may continue provided:

   1. Within 4 hours verify that at least two valves in each high pressure line having a non-functional valve are in, and remain in that mode corresponding to the isolated condition, i.e., manual valves shall be locked in the closed position; motor operated valves shall be placed in the closed position and power supplies deenergized, and

   2. The leakage from the remaining isolation valves in each high pressure line having a valve not meeting the criteria of Table 3.4-1, as listed in Table 3.4-1, shall be determined and recorded daily. The positions of the other valves located in the high pressure line having the leaking valve shall be recorded daily.

   Otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. With any Reactor Coolant System Pressure Isolation Valve leakage greater than 5 gpm, reduce leakage to below 5 gpm, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.2.1 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:

a. Monitoring the containment atmosphere gaseous and/or particulate radioactivity monitor at least once per 12 hours;

b. Monitoring the containment sump inventory at least once per 12 hours;

c. Performance of a Reactor Coolant System water inventory balance at least once per 24 hours; and

d. Monitoring the Reactor Head Flange Leakoff System at least once per 24 hours.

SURVEILLANCE REQUIREMENTS (CONTINUED)

4.4.6.2.2 Each Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1 shall be demonstrated OPERABLE by verifying leakage to be within its limit:*

a. At least once per a refueling shutdown.

b. Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 72 hours or more and if leakage testing has not been performed in the previous 9 months,

c. Prior to returning the valve to service following maintenance, repair or replacement work on the valve, and

The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

* To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

TABLE 3.4-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

VALVE NUMBER		FUNCTION
Unit 3	Unit 4	High-Head Safety Injection Check Valves
3-874A	4-874A	Loop A, hot leg
3-875A	4-875A	cold leg
3-873A	4-873A	cold leg
3-874B	4-874B	Loop B, hot leg
3-875B	4-875B	cold leg
3-873B	4-873B	cold leg
3-875C	4-875C	Loop C, cold leg
3-873C	4-873C	cold leg
		Residual Heat Removal Line Check Valves
3-876A	4-876A	Loop A, cold leg
	4-876E	
3-876B	4-876B	Loop B, cold leg
3-876D	4-876D	
3-876C	4-876C	Loop C, cold leg
3-876E		
MOV3-750	MOV4-750	Loop A, hot leg to RHR
MOV3-751	MOV4-751	Loop A, hot leg to RHR

ACCEPTABLE LEAKAGE LIMITS

1. Leakage rates less than or equal to 1.0 gpm are considered acceptable.

2. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered acceptable provided that the latest measured rate has not exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

3. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered unacceptable if the latest measured rate exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

4. Leakage rates greater than 5.0 gpm are considered unacceptable.

REACTOR COOLANT SYSTEM

3/4.4.7 CHEMISTRY

LIMITING CONDITION FOR OPERATION

3.4.7 The Reactor Coolant System chemistry shall be maintained within the limits specified in Table 3.4-2.

APPLICABILITY: At all times.

ACTION:

MODES 1, 2, 3, and 4:

a. With any one or more chemistry parameter in excess of its Steady-State limit but within its Transient Limit, restore the parameter to within its Steady-State Limit within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; and

b. With any one or more chemistry parameter in excess of its Transient Limit, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours. A Safety Review shall be made prior to reactor startup.

At All Other Times:

With the concentration of either chloride or fluoride in the Reactor Coolant System in excess of its Steady-State Limit for more than 24 hours or in excess of its Transient Limit, reduce the pressurizer pressure to less than or equal to 500 psig, if applicable, and perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation prior to increasing the pressurizer pressure above 500 psig or prior to proceeding to MODE 4.

SURVEILLANCE REQUIREMENT

4.4.7 The Reactor Coolant System chemistry shall be determined to be within the limits by analysis of those parameters at the frequencies specified in Table 4.4-2.

TABLE 3.4-2

REACTOR COOLANT SYSTEM

CHEMISTRY LIMITS

PARAMETER	STEADY-STATE LIMIT	TRANSIENT LIMIT
Dissolved Oxygen*	≤ 0.10 ppm	≤ 1.00 ppm
Chloride	≤ 0.15 ppm	≤ 1.50 ppm
Fluoride	≤ 0.15 ppm	≤ 1.50 ppm

*Limit not applicable with T_{avg} less than or equal to 250 F.

TABLE 4.4-2

REACTOR COOLANT SYSTEM

CHEMISTRY LIMITS SURVEILLANCE REQUIREMENTS

PARAMETER	SAMPLE AND ANALYSIS FREQUENCY
Dissolved Oxygen*	At least 5 times per week not to exceed 72 hours between samples.
Chloride	At least 5 times per week not to exceed 72 hours between samples.
Fluoride	At least 5 times per week not to exceed 72 hours between samples.

*Not required with T_{avg} less than or equal to 250 F

3/4.4.8 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.8 The specific activity of the reactor coolant shall be limited to:

a. Less than or equal to 1 microCurie per gram DOSE EQUIVALENT I-131, and

b. Less than or equal to $100/\bar{E}$ microCuries per gram of gross radioactivity.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

MODES 1, 2 and 3*:

a. With the specific activity of the reactor coolant greater than 1 microCurie per gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure 3.4-1, be in at least HOT STANDBY with T_{avg} less than 500 F within 6 hours;

b. With the gross specific activity of the reactor coolant greater than $100/\bar{E}$ microCuries per gram, be in at least HOT STANDBY with T_{avg} less than 500 F within 6 hours.

MODES 1, 2, 3, 4, and 5:

With the specific activity of the reactor coolant greater than 1 microCurie per gram DOSE EQUIVALENT I-131 or greater than $100/\bar{E}$ microCuries per gram, perform the sampling and analysis requirements of Item 6.a) of Table 4.4-3 until the specific activity of the reactor coolant is restored to within its limits.

SURVEILLANCE REQUIREMENTS

4.4.8 The specific activity of the reactor coolant shall be determined to be within the limits by performance of the sampling and analysis program of Table 4.4-3.

* With T_{avg} greater than or equal to 500 F

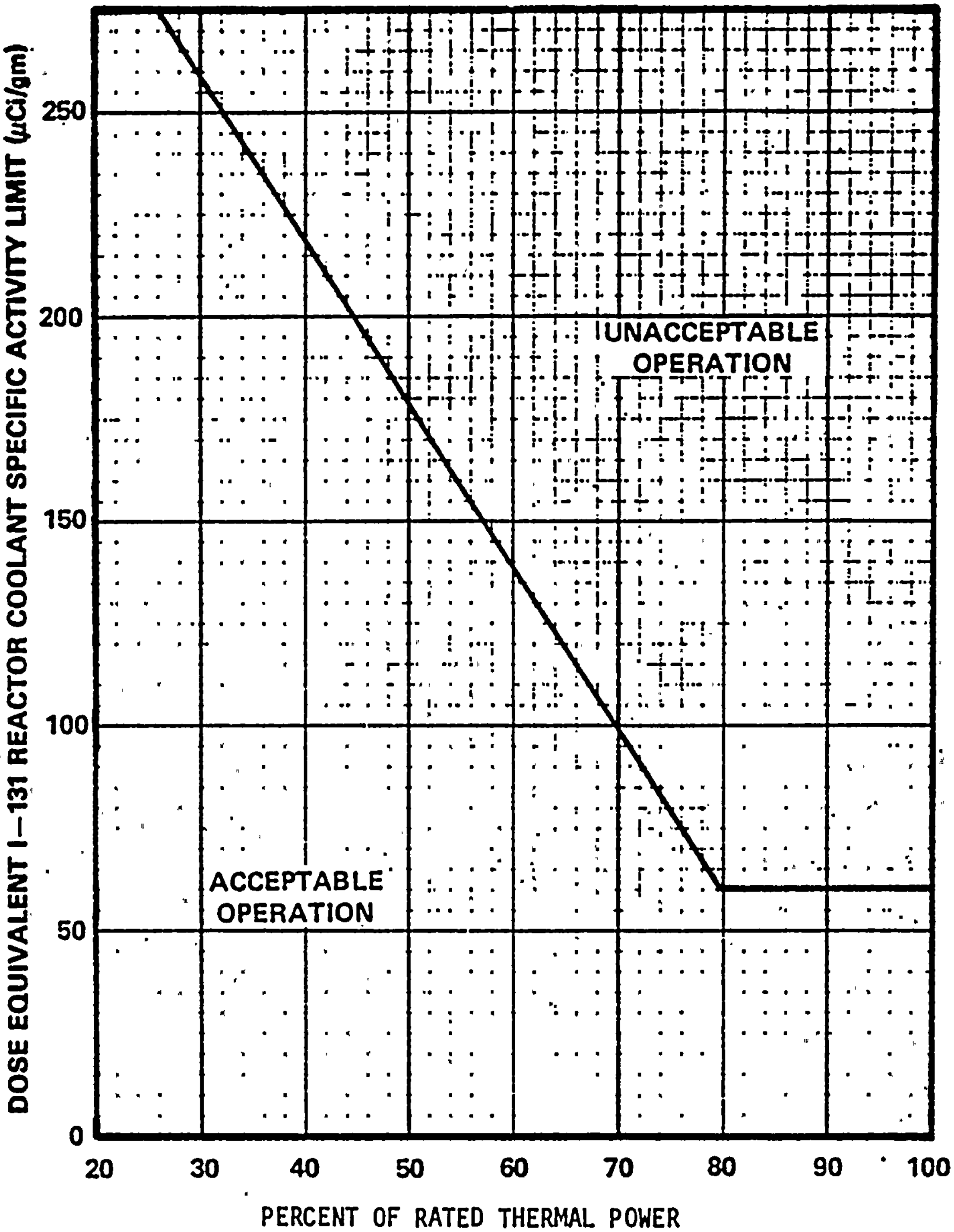


Figure 3.4-1

DOSE EQUIVALENT I-131 REACTOR COOLANT SPECIFIC ACTIVITY LIMIT VERSUS PERCENT OF RATED THERMAL POWER WITH THE REACTOR COOLANT SPECIFIC ACTIVITY >1 µCi/gram DOSE EQUIVALENT I-131.

TABLE 4.4-3

REACTOR COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	MODES IN WHICH SAMPLE AND ANALYSIS REQUIRED
1. Gross Radioactivity Determination	At least once per 72 hours.	1, 2, 3, 4
2. Tritium Activity Determination	1 per 7 days	1, 2, 3, 4
3. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	1 per 14 days.	1
4. Radiochemical Isotopic Determination including Gaseous Activity	Monthly	1, 2, 3, 4
5. Radiochemical for $\bar{E}$ Determination	1 per 6 months*	1
6. Isotopic Analysis for Iodine Including I-131, I-133, and I-135	a) Once per 4 hours, whenever the specific activity exceeds 1 μCi/gram DOSE EQUIVALENT I-131 or 100/$\bar{E}$ μCi/gram of gross radioactivity, and	1**,2**,3**,4**,5**
	b) One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15% of the RATED THERMAL POWER within a 1-hour period.	1, 2, 3

TABLE 4.4-3 (Continued)

TABLE NOTATIONS

* Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since reactor was last subcritical for 48 hours or longer.

** Until the specific activity of the Reactor Coolant System is restored within its limits.

REACTOR COOLANT SYSTEM

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.9.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2 and 3.4-3 for Unit 3 and Figures 3.4-4 and 3.4-5 for Unit 4 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

a. A maximum heatup rate of 100 F in any 1-hour period.

b. A maximum cooldown rate of 100 F in any 1-hour period, and

c. A maximum temperature change of less than or equal to 5 F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

APPLICABILITY: At all times

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{avg} and pressure to less than 200 F and 500 psig, respectively, within the following 30 hours.

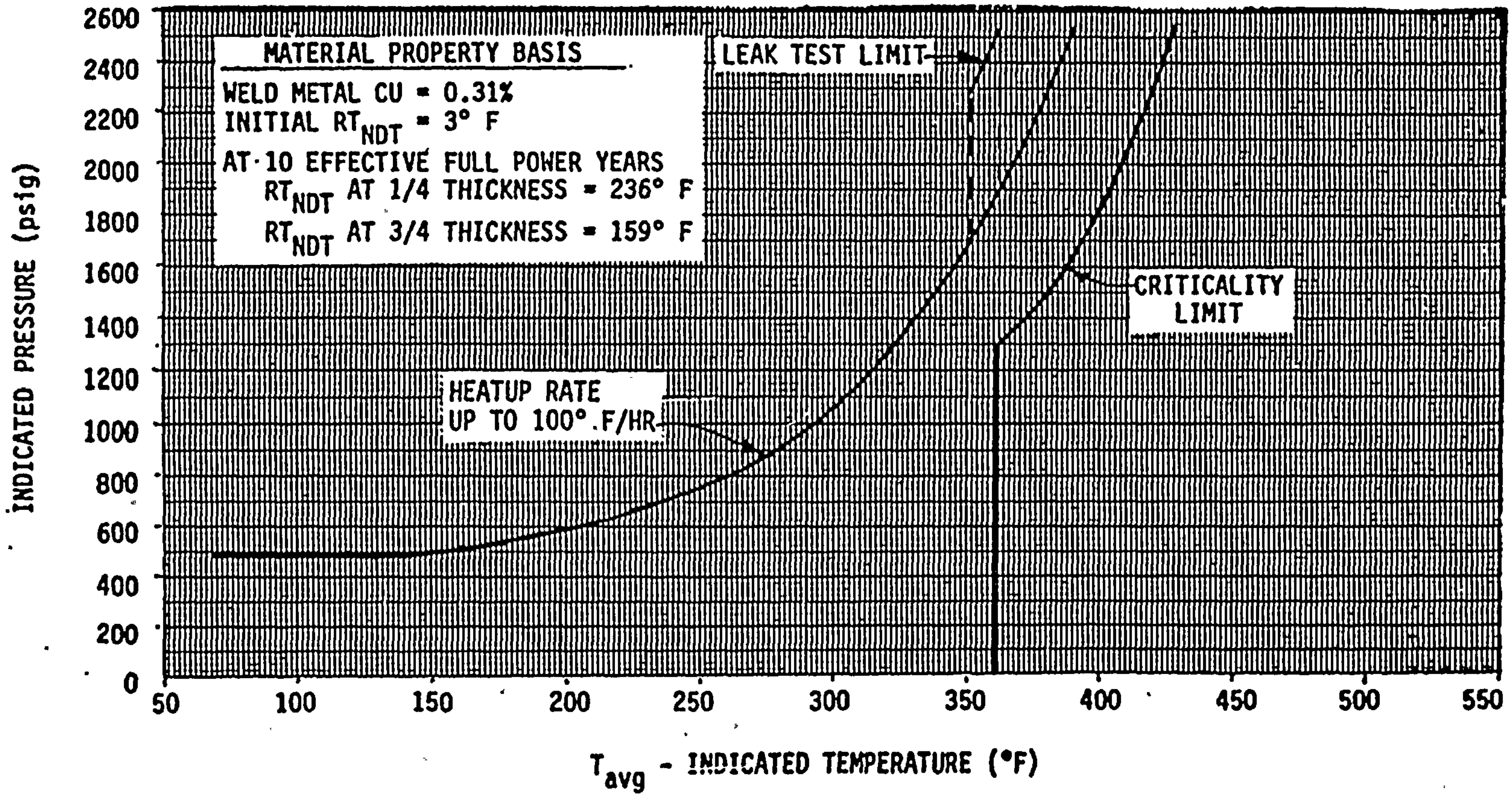
SURVEILLANCE REQUIREMENTS

4.4.9.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

4.4.9.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, as required by 10 CFR Part 50, Appendix H, in accordance with the schedule in Table 4.4-4. The results of these examinations shall be used to update Figures 3.4-2 to 3.4-5.

REACTOR COOLANT SYSTEM HEATUP LIMITATIONS - APPLICABLE TO 5 TO 10 EFPY

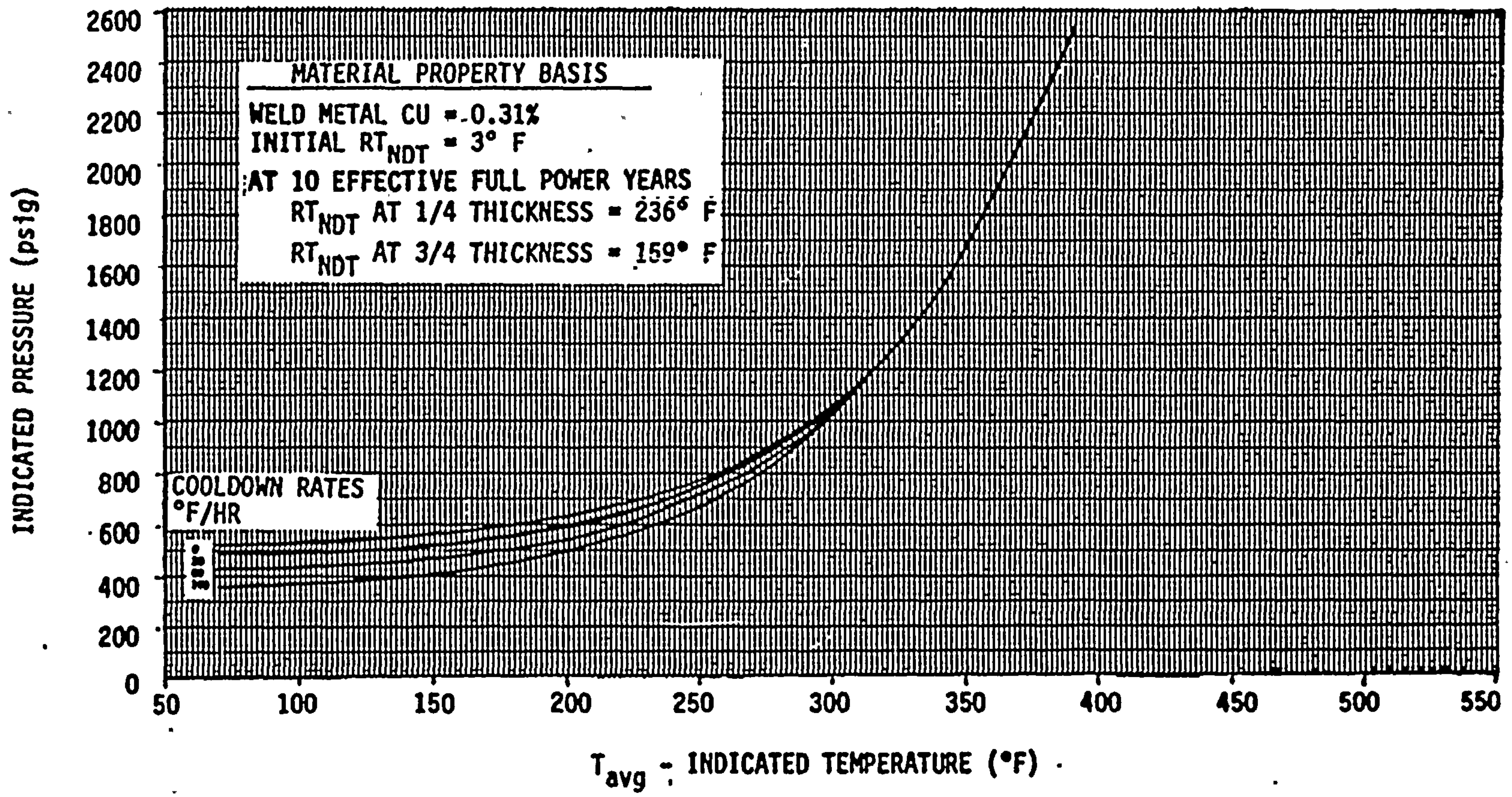
FIGURE 3.4-2



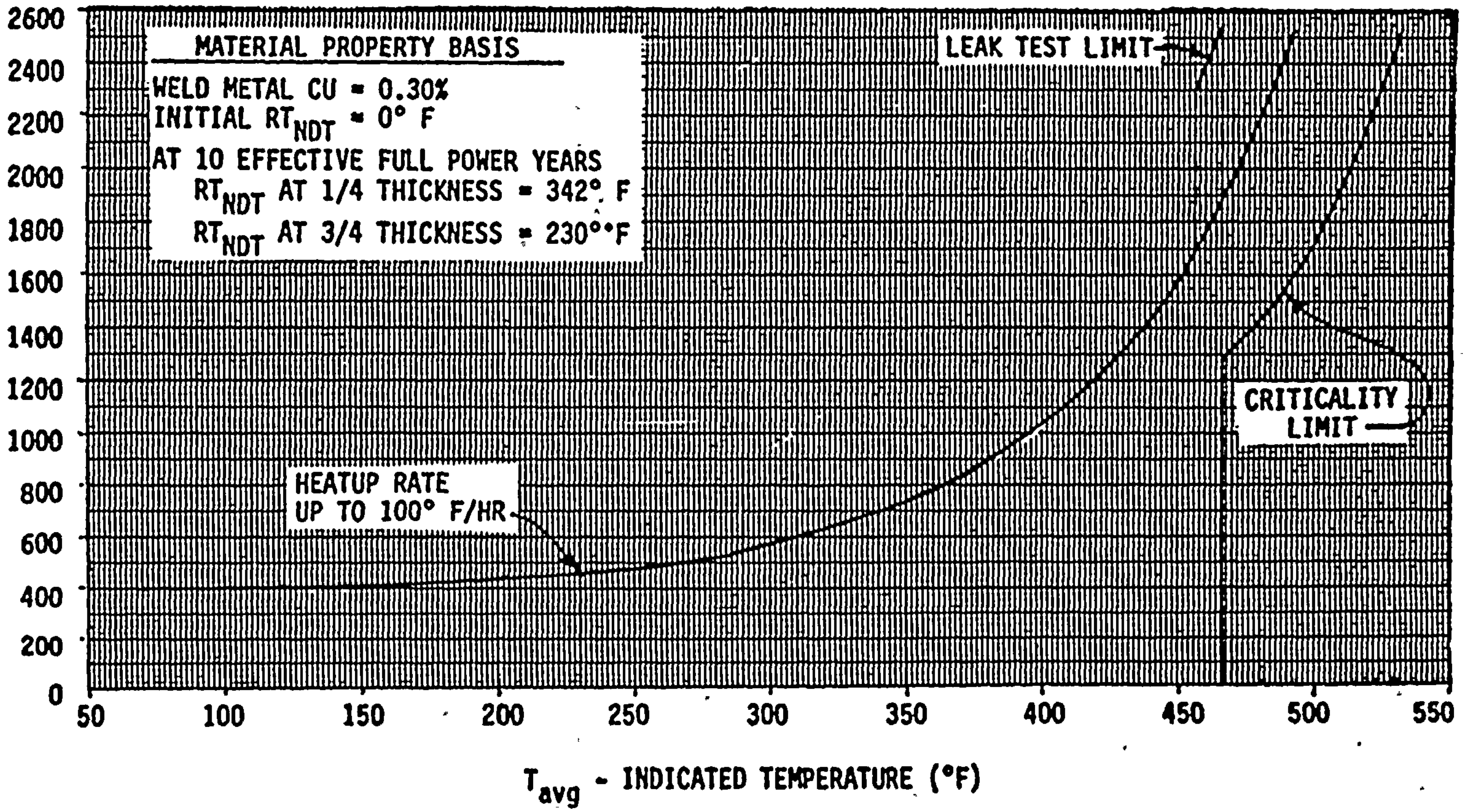
TURKEY POINT UNIT 3 REACTOR COOLANT HEATUP LIMITATIONS APPLICABLE FOR PERIODS FROM 5 TO 10 EFFECTIVE FULL POWER YEARS.

REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS - APPLICABLE TO 5 TO 10 EFPY

FIGURE 3.4-3



TURKEY POINT UNIT 3 REACTOR COOLANT COOLDOWN LIMITATIONS APPLICABLE FOR PERIODS FROM 5 TO 10 EFFECTIVE FULL POWER YEARS.



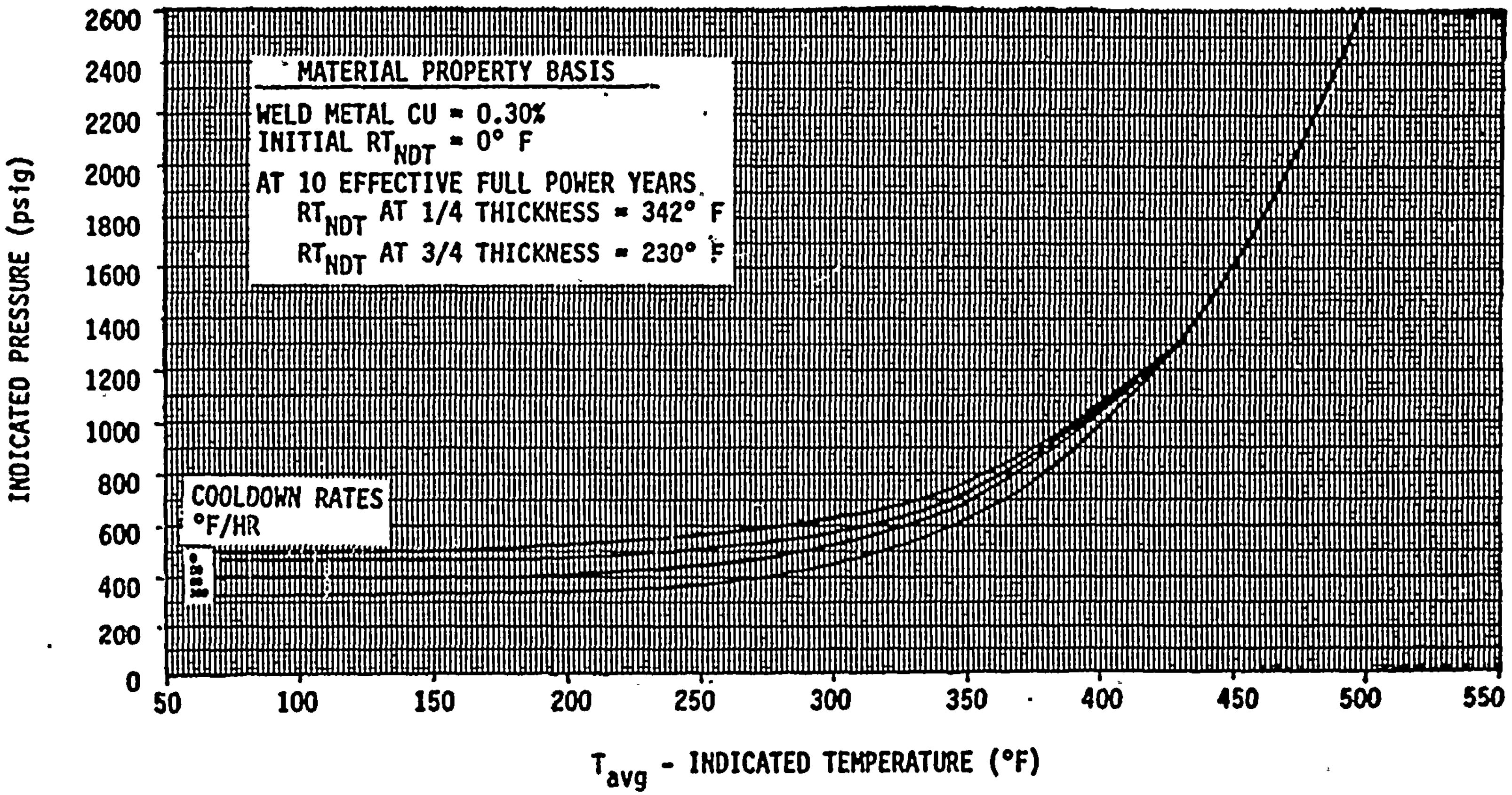
TURKEY POINT UNIT 4 REACTOR COOLANT HEATUP LIMITATIONS APPLICABLE FOR PERIODS FROM 5 TO 10 EFFECTIVE FULL POWER YEARS.

FIGURE 3.4-4

REACTOR COOLANT SYSTEM HEATUP LIMITATIONS - APPLICABLE TO 5 TO 10 EFPY

FIGURE 3.4-5

REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS - APPLICABLE TO 5 TO 10 EFPY



TURKEY POINT UNIT 4 REACTOR COOLANT COOLDOWN LIMITATIONS APPLICABLE FOR PERIODS FROM 5 TO 10 EFFECTIVE FULL POWER YEARS.

TABLE 4.4-4

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM - WITHDRAWAL SCHEDULE

UNIT 3

CAPSULE	WITHDRAWAL TIME
U	Standby
V	12 years
W	Standby
X	33 years
Y	Standby
Z	Standby

UNIT 4

CAPSULE	WITHDRAWAL TIME
U	Standby
V	24 years
W	Standby
X	Standby
Y	Standby
Z	Standby

PRESSURIZER - PRESSURE/TEMPERATURE LIMITS

LIMITING CONDITION FOR OPERATION

3.4.9.2 The pressurizer temperature shall be limited to:

a. A maximum heatup rate of 100 F in any 1-hour period,

b. A maximum cooldown rate of 200 F in any 1-hour period, and

c. A maximum spray water temperature differential of 320 F.

APPLICABILITY: At all times.

ACTION:

With the pressurizer temperature limits in excess of any of the above limits, restore the temperature to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the pressurizer; determine that the pressurizer remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the pressurizer pressure to less than 500 psig within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.2 The pressurizer temperatures shall be determined to be within the limits at least once per 30 minutes during system heatup or cooldown. The spray water temperature differential shall be determined to be within the limit at least once per 12 hours during auxiliary spray operation.

REACTOR COOLANT SYSTEM

OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.9.3 Valves MOV-843A, MOV-843B, and MOV-869 shall be closed and their breakers racked out, and at least one of the following Overpressure Protection Systems shall be OPERABLE **,***:

a. Two power-operated relief valves (PORVs) with a lift setting of less than or equal to 415 psig ± 15 psig whenever RCS cold leg temperature is 275 F or less, or

b. The Reactor Coolant System (RCS) depressurized with an RCS vent of greater than or equal to 2.20 square inches.

APPLICABILITY: MODES 3*, 4, 5 and 6

ACTION

In MODE 3 with T_{avg} an or equal to 380 F and MODE 4, 5 and 6 with the reactor vessel head on, if any of the MOV valves listed in 3.4.9.3 are found to be open, perform at least one of the following within the next 8 hours:

a. Block the corresponding flow path to the reactor vessel, or

b. Close the valve, or

c. Depressurize and vent the RCS through an opening with an area of at least 2.20 square inches, or

d. Verify at least one pressurizer power operated relief valve is maintained open.

In MODE 4 when the temperature of any RCS cold leg is less than or equal to 275 F or in MODE 5 and MODE 6 with the reactor vessel head on,

a. With one PORV inoperable, restore the inoperable PORV to OPERABLE status within 7 days or depressurize and vent the RCS through at least a 2.20 square inch vent within the next 8 hours.

---

* When T_{avg} is less than or equal to 380 F

** MOV-843A, MOV-843B and MOV-869 may be energized and open for maintance, testing and accumulator fill provided the affected flow path is blocked by closure of an alternate valve(s). Surveillance requirement 4.4.9.3.3 would then apply to the alternate valve(s).

*** MOV-843A, MOV-843B and MOV-869 may be open to establish a flow path to the RCS for the Safety Injection System full flow pump and valve test provided the conditions of Specification 3.4.9.3.b are established.

OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION (CONTINUED)

b. With both PORVs inoperable, depressurize and vent the RCS through at least a 2.20 square inch vent within 24 hours.

c. In the event either the PORVs or the RCS vent(s) are used to mitigate an RCS pressure transient that exceeds the pressure and/or temperature limits of Figure(s) 3.4-2,3,4 or 5, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs or RCS vent(s) on the transient, and any corrective action necessary to prevent recurrence.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.9.3.1 Each PORV shall be demonstrated OPERABLE by:

a. Performance of an ANALOG CHANNEL OPERATIONAL TEST on the PORV actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the PORV is required OPERABLE and at least once per 31 days thereafter when the PORV is required OPERABLE.

b. Performance of a CHANNEL CALIBRATION on the PORV actuation channel at least once each refueling; and

c. Verifying the PORV isolation valve is open at least once per 72 hours when the PORV is being used for overpressure protection.

d. While the PORVs are required to be OPERABLE, the backup air supply shall be verified OPERABLE at least once per 24 hours.

4.4.9.3.2 The RCS vent(s) shall be verified to be open at least once per 12 hours* when the vent(s) is being used for overpressure protection.

4.4.9.3.3 Verify valves MOV-843A, MOV-843B and MOV-869 are closed at least once per 24 hours when the valves are required to be closed by Specification 3.4.9.3.

* Except when the vent pathway is provided with a valve which is locked, sealed, or otherwise secured in the open position, then verify these valves open at least once per 31 days.

REACTOR COOLANT SYSTEM

3/4.4.10 STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.4.10 The structural integrity of ASME Code Class 1, 2, and 3 components shall be maintained in accordance with Specification 4.4.10.

APPLICABILITY: All MODES.

ACTION:

a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50 F above the minimum temperature required by NDT considerations.

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200 F.

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) from service.

d. The provisions of Specification 3.0.4 are not applicable.

REACTOR COOLANT SYSTEM

3/4.4.10 STRUCTURAL INTEGRITY

SURVEILLANCE REQUIREMENTS

4.4.10 Requirements of Specification 4.0.5 shall be met including the following specific requirements.

a. Nondestructive inspections required by Specification 4.0.5 shall be performed as specified.

b. Reactor Coolant System integrity shall be demonstrated as follows after the system is closed following normal opening, modification or repair.

   1) When the Reactor Coolant System is closed, the system will be leak tested at not less than 2335 psig while meeting NDTT requirements for temperature.

   2) When Reactor Coolant System modifications or repairs have been made which involved new strength welds on components greater than 2 in. diameter, the new welds will receive both a surface and 100% volumetric examination.

   3) When Reactor Coolant System modifications or repairs have been made which involve new strength welds on components 2 in. diameter or smaller, the new welds will receive a surface examination.

3/4.4.11 REACTOR COOLANT SYSTEM VENTS

LIMITING CONDITION FOR OPERATION

3.4.11 At least one Reactor Coolant System vent path consisting of at least two vent valves in series powered from emergency busses shall be OPERABLE and closed at each of the following locations:

a. Reactor vessel head,

b. Pressurizer steam space.

APPLICABILITY: MODES 1, 2, 3, and 4

ACTION:

a. With one of the above Reactor Coolant System vent paths inoperable, STARTUP and/or POWER OPERATION may continue provided the inoperable vent path is maintained closed with power removed from the valve actuator of all the vent valves in the inoperable vent path; restore the inoperable vent path to OPERABLE status within 30 days, or, be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With two or more Reactor Coolant System vent paths inoperable; maintain the inoperable vent path closed with power removed from the valve actuators of all the vent valves in the inoperable vent paths, and restore at least one of the vent paths to OPERABLE status within 72 hours or be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.11.1 Each Reactor Coolant System vent path downstream vent valve not required to be closed by ACTION a. or b., above, shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel from the control room.

4.4.11.2 Each Reactor Coolant System vent path shall be demonstrated OPERABLE at least once per refueling by:

a. Verifying all manual isolation valves in each vent path are locked in the open position.

b. Cycling each vent valve through at least one complete cycle of full travel from the control room, and

c. Verifying flow through the Reactor Coolant System vent paths during COLD SHUTDOWN or REFUELING.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1 Each Reactor Coolant System (RCS) accumulator shall be OPERABLE with:

a. The isolation valve open and its circuit breaker locked open,

b. A contained borated water volume of between 6545 and 6664 gallons,

c. A boron concentration of between 1950 and 2350 ppm, and

d. A nitrogen cover-pressure of between 600 and 675 psig.

APPLICABILITY: MODES 1, 2 and 3*.

ACTION:

a. With one accumulator inoperable, except as a result of a closed isolation valve, restore the inoperable accumulator to OPERABLE status within 4 hours or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

b. With one accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.1 Each accumulator shall be demonstrated OPERABLE:

a. At least once per shift or 12 hours, whichever is more limiting, by:

1) Verifying the contained borated water volume and nitrogen cover-pressure in the tanks, and

2) Verify that each accumulator isolation valve is open by control room indication (power maybe restored to the valve operator to perform this surveillance if the redundant indicator is inoperable).

---

* Pressurizer pressure above 1000 psig

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days and within 6 hours after each solution volume increase of greater than or equal to 1% of tank volume by verifying the boron concentration of the accumulator solution; and

c. At least once per 31 days when the RCS pressure is above 1000 psig, by verifying that the power to the isolation valve operator is disconnected by a locked open breaker.

d. During each refueling shutdown, accumulator check valves shall be checked for operability.

4.5.1.2 Each accumulator water level and pressure channel shall be demonstrated OPERABLE:

a. At least once per 31 days by the performance of an ANALOG CHANNEL OPERATIONAL TEST, and

b. During each refueling by the performance of a CHANNEL CALIBRATION.

3/4.5.2 ECCS SUBSYSTEMS - T_{avg} GREATER THAN OR EQUAL TO 350 F

LIMITING CONDITION FOR OPERATION

3.5.2 The following Emergency Core Cooling System (ECCS) equipment and flow paths shall be OPERABLE:

a. Four OPERABLE Safety Injection pumps,

b. Two OPERABLE RHR heat exchangers,

c. Two OPERABLE RHR pumps,

d. One OPERABLE flow path** capable of taking suction from the refueling water storage tank and delivering water to the RCS via the SI and RHR pumps, and

e. Two OPERABLE flow paths capable of taking suction from the containment sump and delivering water to the RCS via the RHR pumps.

APPLICABILITY: MODES 1, 2, and 3.*

ACTION:

a. With any one of the required ECCS component or flow paths inoperable, except for inoperable Safety Injection Pump(s), restore the inoperable component or flow path to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one Safety Injection Pump inoperable, restore the pump to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With two Safety Injection Pumps inoperable, restore one of the two inoperable pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

---

* Specification 3.5.2.a and 3.5.2.d (for SI pumps only) are not applicable at RCS cold leg temperature of $\leq$380 F due to overpressurization protection requirements that isolate High Head SI from RCS.

** During refueling operation of one unit, all OPERABLE High Head SI pumps shall be aligned to the RWST of the operating unit.

SURVEILLANCE REQUIREMENTS

4.5.2 The ECCS equipment and flow paths shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying(by control room indication) that the following valves are in the indicated positions with power to the valve operators removed*:

Valve Number	Valve Function	Valve Position
864 A and B	Supply From RWST to ECCS	Open
862 A and B	RWST Supply to RHR pumps	Open
863 A and B	RHR Recirculation	Closed
866 A and B	H.H.S.I. to Hot Legs	Closed
HCV-758	RHR HX Outlet	Open

*Air supply to HCV-758 shall be verified shut off once per 31 days.

To permit temporary operation of these valves for surveillance or maintenance purposes, power may be restored to these valves for a period not to exceed 24 hours.

b. At least once per 31 days by:

1) Verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position, and

2) Verifying that power is available to flow path components that require power for flow path operability.

c. At least once per 92 days by:

1) Verifying that the ECCS piping is full of water by venting as available ECCS pump casings and accessible discharge piping, and

2) Verifying RHR and Safety Injection pumps start and reach their required head for normal or recirculation flow, whichever is applicable to the operating conditions pursuant to Specification 4.0.5; the instruments and visual observations shall indicate proper functioning. Test operation shall be for at least 15 minutes.

d. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment sump and cause restriction of the pump suctions during LOCA conditions. This visual inspection shall be performed:

1) For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and

2) Of the areas affected within containment at the completion of each containment entry when CONTAINMENT INTEGRITY is established.

e. At least once per refueling by:

1) Verifying automatic isolation and interlock action of the RHR system from the Reactor Coolant system by ensuring that with a simulated or actual Reactor Coolant System pressure signal greater than 525 psig, the interlocks cause the valves to automatically close and prevent the valves from being opened.

2) Verifying correct interlock action to ensure that the RWST is isolated from the RHR System during RHR System operation and to ensure that the RHR System cannot be pressurized from the Reactor Coolant System unless the above RWST Isolation Valves are closed.

3) A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or abnormal corrosion.

f. At least once per refueling by:

1) Verifying that each automatic valve in the flow path actuates to its correct position on a Safety Injection actuation test signal, and

2) Verifying that each of the following pumps start automatically upon receipt of a Safety Injection actuation test signal:

a) Safety Injection pump, and

b) RHR pump.

SURVEILLANCE REQUIREMENTS (Continued)

g. Prior to initiating maintenance on an inoperable RHR pump or two inoperable SI pumps, demonstrate operability of the unaffected RHR or SI pump(s) by performing requirements of Specification 4.5.2.c above.

h. Prior to initiating maintenance on any inoperable valve, demonstrate operability of valve(s) that provide duplicate flow path function to the inoperable valve(s).

3/4.5.3 ECCS SUBSYSTEMS - T_{avg} LESS THAN 350 F

LIMITING CONDITION FOR OPERATION

3.5.3 As a minimum, the following ECCS equipment and flow path shall be OPERABLE:

a. One OPERABLE RHR heat exchanger,

b. One OPERABLE RHR pump, and

c. An OPERABLE flow path capable of taking suction from the refueling water storage tank upon being manually realigned and transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODE 4.

ACTION:

a. With no OPERABLE ECCS flow path from the refueling water storage tank, restore at least one ECCS flow path to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 20 hours.

b. With either the residual heat removal heat exchanger or RHR pump inoperable, restore the required equipment to OPERABLE status or maintain the Reactor Coolant System T_{avg} less than 350 F by use of alternate heat removal methods.

SURVEILLANCE REQUIREMENTS

4.5.3 The ECCS equipment and flow path shall be demonstrated OPERABLE per the applicable requirements of Specification 4.5.2.

EMERGENCY CORE COOLING SYSTEMS

3/4.5.4 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.4 The refueling water storage tank (RWST) shall be OPERABLE with:

a. A minimum contained borated water volume of 320,000 gallons,

b. A minimum boron concentration of 1950 ppm of boron,

c. A minimum solution temperature of 39 F, and

d. A maximum solution temperature of 100 F.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the RWST inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.4 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the contained borated water volume in the tank, and

2) Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWST temperature* is within limits whenever the outside air temperature is less than 39 F or greater than 100 F.

* Portable instrumentation may be used to monitor the RWST temperature.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.**

APPLICABILITY: MODES 1, 2, 3, 4

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

** Exception may be taken under Administrative Controls for opening of certain valves and airlocks necessary to perform surveillance or testing requirements.

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that all penetrations* not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Table 3.6-1 of Specification 3.6.4.1;

b. By verifying that each containment air lock is in compliance with the requirements of Specification 3.6.1.3; and

c. After each closing of each penetration subject to Type B testing, except the containment air locks, if opened following a Type A or B test, by leak rate testing the seal with gas at a pressure not less than 50 psig, and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Specification 4.6.1.2.3. for all other Type B and C penetrations, the combined leakage rate is less than 0.60 L_a.

---

* Except valves, blind flanges, and deactivated automatic valves which are located inside the containment and are locked, sealed or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that such verification need not be performed more often than once per 92 days.

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of:

1) Less than or equal to L_a, 0.25% by weight of the containment air per 24 hours at P_a, 49.9 psig, or

2) Less than or equal to L_t, 0.177% by weight of the containment air per 24 hours at a reduced pressure of P_t, 25 psig.

b. A combined leakage rate of less than 0.60 L_a for all penetrations and valves subject to Type B and C tests, when pressurized to 50 psig.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With either the measured overall integrated containment leakage rate exceeding 0.75 L_a or 0.75 L_t, as applicable, or the measured combined leakage rate for all penetrations and valves subject to Types B and C tests exceeding 0.60 L_a, restore the overall integrated leakage rate to less than 0.75 L_a or less than 0.75 L_t, as applicable, and the combined leakage rate for all penetrations subject to Type B and C tests to less than 0.60 L_a prior to increasing the Reactor Coolant System temperature above 200 F.

SURVEILLANCE REQUIREMENTS

4.6.1.2.1 The containment leakage rates shall be demonstrated and determined in conformance with the criteria specified in Appendix J of 10CFR Part 50 (1985) using the methods and provisions of ANSI N45.4-[1972].

4.6.1.2.2 Three Type A tests (Overall Integrated Containment Leakage Rate) shall be conducted at 40 ± 10 month intervals during shutdown at a pressure not less than either P_a, 49.9 psig, or at P_t, 25 psig, during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection;

SURVEILLANCE REQUIREMENTS (CONTINUED)

4.6.1.2.3 Type B and C tests shall be conducted with gas at a pressure not less than 50 psig, during each reactor shutdown for refueling, but in no case at intervals greater than 24 months except for tests involving:

a. Air Locks,*

b. Purge Supply and exhaust isolation valves,*

c. Equipment access opening* which shall be tested at least once every 12 months and after each use.

4.6.1.2.4 Air locks shall be tested and demonstrated OPERABLE by the requirements of Specification 4.6.1.3;

4.6.1.2.5 Purge supply and exhaust isolation valves shall be tested and demonstrated OPERABLE by the requirements of Specification 4.6.1.7.2;

4.6.1.2.6 The provisions of Specification 4.0.2 are not applicable.

* The 0.60 L_a leakage limit of Specification 3.6.1.2.b shall not be exceeded when the measured leakage rates for the components are added to the previously determined total for all other penetrations subject to Type B and C tests.

CONTAINMENT SYSTEMS

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with:

a. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed, and

b. An overall air lock leakage rate of less than or equal to 0.2 L_a at 50 psig.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With one containment air lock door inoperable:

1. Maintain at least the OPERABLE air lock door closed and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed;

2. Operation may then continue until performance of the next required overall air lock leakage test provided that the OPERABLE air lock door is verified to be locked closed at least once per 31 days;

3. Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; and

4. The provisions of Specification 3.0.4 are not applicable.

b. With the containment air lock inoperable, except as the result of an inoperable air lock door, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. Within 72 hours following each closing, except when the air lock is being used for multiple entries, then at least once per 72 hours, by verifying that seals have not been damaged and have seated properly by vacuum testing the volume between the door seals in accordance with approved plant procedures.

b. By conducting overall air lock leakage tests at not less than 50 psig, and verifying the overall air lock leakage rate is within its limit:

1) At least once per 6 months,*

c. At least once per 6 months by verifying that only one door in each air lock can be opened at a time.

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* The provisions of Specification 4.0.2 are not applicable.

CONTAINMENT SYSTEMS

INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.4 Primary containment internal pressure shall be maintained between -2 and +3 psig.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the containment internal pressure outside of the limits above, restore the internal pressure to within the limits within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.4 The primary containment internal pressure shall be determined to be within the limits at least once per 12 hours.

AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.5 Primary containment average air temperature shall not exceed 120 F.

APPLICABILITY: Modes 1, 2, 3 and 4.

ACTION:

With the indicated containment average air temperature greater than 120 F within 8 hours confirm the temperature using an alternate method in at least three locations on the 58' elevation. With confirmed containment average air temperature greater than 120 F, reduce the average air temperature to within the limit within 72 hours, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.5 The primary containment average air temperature shall be the arithmetical average of the temperatures at the following locations and shall be determined at least once per 24 hours:

Locations*:

Containment Air Temperature Monitors

a. TE-6700

b. TE-6701

c. TE-6702

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* With any of the above temperature monitors out of service use monitor(s) TE-1483, TE-1484 , TE-1487 or TE-1488.

CONTAINMENT VESSEL STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.6 The structural integrity of the containment vessel shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.6.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With more than one tendon with an observed lift-off force between the predicted lower limit and 90% of the predicted lower limit or with one tendon below 90% of the predicted lower limit, restore the tendon(s) to the required level of integrity within 15 days and perform an engineering evaluation of the containment and provide a Special Report to the Commission within 30 days in accordance with Specification 6.9.2 or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any abnormal degradation of the structural integrity other than ACTION a. at a level below the acceptance criteria of Specification 4.6.1.6, restore the containment vessel to the required level of integrity within 72 hours and perform an engineering evaluation of the containment and provide a Special Report to the Commission within 15 days in accordance with Specification 6.9.2 or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.6.1 Containment Vessel Tendons. The containment vessel tendons' structural integrity shall be demonstrated every fifth year from the date of the Structural Integrity Test. The tendons' structural integrity shall be demonstrated by:

a. Determining that a representative sample of at least 10 tendons (3 dome, 3 vertical, and 4 hoop) each have an observed lift-off force within predicted limits for each. For each subsequent inspection one tendon from each group may be kept unchanged to develop a history and to correlate the observed data. If the observed lift-off force of any one tendon in the original sample population lies between the predicted lower limit and 90% of the predicted lower limit, two tendons, one on each side of this tendon should be checked for their lift-off forces. If both of these adjacent tendons are found to be within their predicted limits, all three tendons should be restored to the required level of integrity. This single deficiency may be considered unique and acceptable.

b. Performing tendon detensioning, inspections, and material tests on a previously stressed tendon from each group (dome, vertical, and hoop). A randomly selected accessible tendon from each group shall be completely detensioned in order to identify broken or damaged wires and determining that over the entire length of the removed wire or strand that:

   1) The tendon wires or strands are free of corrosion, cracks, and damage,

   2) There are no changes in the presence or physical appearance of the sheathing filler-grease, and

   3) A minimum tensile strength of 240,000 psi (guaranteed ultimate strength of the tendon material) for at least three wire or strand samples (one from each end and one at mid-length) cut from each removed wire or strand. Failure of any one of the wire or strand samples to meet the minimum tensile strength test is evidence of abnormal degradation of the containment vessel structure.

c. Performing tendon retensioning of those tendons detensioned for inspection to their observed lift-off force with a tolerance limit of ± 6%. During retensioning of these tendons, the changes in load and elongation should be measured simultaneously at a minimum of three approximately equally spaced levels of force between zero and the seating force. If the elongation corresponding to a specific load differs by more than 5% from that recorded during installation, an investigation should be made to ensure that the difference is not related to wire failures or slip of wires in anchorages;

d. Assuring the observed lift-off stresses exceed the average minimum design value given below, which are adjusted to account for elastic losses; and

Dome	133	ksi
Vertical	133	ksi
Hoop	133	ksi

e. Verifying the OPERABILITY of the sheathing filler grease by:

   1) Minimum grease coverage exists for the different parts of the anchorage systems, and

   2) The chemical properties of the filler material are within the tolerance limits as specified by the manufacturer.

4.6.1.6.2 End Anchorages and Adjacent Concrete Surfaces. The structural integrity of the end anchorages of all tendons inspected pursuant to Specification 4.6.1.6.1 and the adjacent concrete surfaces shall be demonstrated by determining through inspection that no apparent changes have occurred in the visual appearance of the end anchorage or the concrete crack patterns adjacent to the end anchorages. Inspections of the concrete shall be performed during the Type A containment leakage rate tests (reference Specification 4.6.1.2) while the containment vessel is at its maximum test pressure.

4.6.1.6.3 Containment Vessel Surfaces. The structural integrity of the exposed accessible interior and exterior surfaces of the containment vessel, including the liner plate, shall be determined during the shutdown for each Type A containment leakage rate test (reference Specification 4.6.1.2) by a visual inspection of these surfaces. This inspection shall be performed prior to the Type A containment leakage rate test to verify no apparent changes in appearance or other abnormal degradation. The first inspection performed will form the baseline for future surveillances.

CONTAINMENT SYSTEMS

CONTAINMENT VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.7 Each containment purge supply (48 inch) and exhaust (54 inch) isolation valves shall be OPERABLE* **.

APPLICABILITY: MODES 1, 2, 3**, AND 4**

ACTION:

THE FOLLOWING ACTIONS APPLY FOR MODE 1, OR 2 OPERATION

1. With the purge supply and/or exhaust isolation valve(s) open wider than 33 or 30 degrees, respectively, within 1 hour close the valve(s) to less than or equal to 33 or 30 degrees, respectively, or close the valve(s), or be in HOT STANDBY within next 6 hours and in COLD SHUTDOWN within the subsequent 30 hours.

2. With the purge supply and/or exhaust isolation valve(s) open for more than 200 hours per unit per calendar year, close the open valve(s) within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the subsequent 30 hours.

3. With the purge supply and/or exhaust isolation valve(s) having a measured leakage rate in excess of the limits of Specification 4.6.1.7.2 and the containment combined leakage for Type B and C penetrations is less than 0.60 L_a, restore the valve leakage within its limit during the next COLD SHUTDOWN condition of 72 hours or longer.

THE FOLLOWING ACTIONS APPLY FOR MODE 3, OR 4 OPERATION

1. With the purge supply and/or exhaust isolation valve(s) open wider than 33 or 30 degrees, respectively, within 1 hour close the valve(s), to less than or equal to 33 or 30 degrees, respectively, or close the valve(s), or be in COLD SHUTDOWN within the next 30 hours.

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* Each containment purge supply and exhaust isolation valve (s) opening shall be limited to less than or equal to a 33 or 30 degree opening, respectively.

** Each containment purge supply and exhaust isolation valve (s) total opening time during a calendar year shall not exceed 200 hours unless a relief has been requested and received from the Commission. In the interim 200 hours limit applies in MODES 1 and 2 operation and is not applicable in MODES 3 and 4 operation.

CONTAINMENT VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION (CONTINUED)

ACTION (Continued):

2. With the purge supply and/or exhaust isolation valve(s) having a measured leakage rate in excess of the limits of Specification 4.6.1.7.2 and the combined leakage for Type B and C penetrations is less than 0.60 L_a, restore the valve leakage within its limit during the next COLD SHUTDOWN condition of 72 hours or longer.

SURVEILLANCE REQUIREMENTS

4.6.1.7.1 The cumulative time that the purge supply and exhaust valves have been open in MODES 1, and 2 during the calendar year shall be determined at least once per 7 days.

4.6.1.7.2 At least once per refueling interval each containment purge supply and exhaust isolation valve shall be demonstrated OPERABLE by verifying that the measured leakage rate is less than or equal to 0.20 L_a when pressurized to P_a.

4.6.1.7.3 At least once per 24 hours verify that each containment purge supply and exhaust isolation valve is closed, or, is open no more than 33 or 30 degrees, respectively.

CONTAINMENT SYSTEMS

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two independent Containment Spray Systems shall be OPERABLE with each Spray System capable of taking suction from the RWST.

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

a. With one Containment Spray System inoperable, restore the inoperable Spray System to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With two Containment Spray Systems inoperable, restore at least one Spray System to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore both Spray Systems to OPERABLE status within 72 hours of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each Containment Spray System shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct positon; and that power is available to flow path components that require power for operation;

b. Each Containment Spray Pump shall be demonstrated OPERABLE pursuant to Specification 4.0.5. The test operation shall be for at least 15 minutes.

CONTAINMENT SPRAY SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

c. At each refueling by:

   1) Verifying that each automatic valve in the flow path actuates to its correct position on a containment spray actuation test signal, and

   2) Verifying that each spray pump starts automatically on a containment spray actuation test signal. The manual isolation valves in the spray lines at the containment shall be locked closed for the performance of these tests.

d. At least once per 5 years by performing an air or smoke flow test through each spray header and verify each spray nozzle is unobstructed.

e. By performing surveillance requirements 4.6.2.1.a for the redundant flow path prior to initiating maintenance on valve(s) in the affected flow path being declared inoperable.

f. By performing surveillance requirement 4.6.2.1.b for the redundant spray pump prior to initiating maintenance on the affected pump being declared inoperable.

CONTAINMENT SYSTEMS

CONTAINMENT COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.2 Three independent emergency containment cooling units shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

a. With one of the above required containment cooling units inoperable, restore the inoperable cooling unit to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With two or more of the above required containment cooling units inoperable, restore at least two cooling units to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore all of the above required cooling units to OPERABLE status within 72 hours of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.2 Each emergency containment cooling fan shall be demonstrated OPERABLE:

a. At least once per 31 days by:

   1) Starting each cooler fan from the control room and verifying that each fan motor reaches the nominal operating current for the test conditions and operates for at least 15 minutes, and

   2) Verifying a cooling water flow rate of greater than or equal to 2000 gpm to each cooler.

b. At least once per each refueling by verifying that each fan starts automatically on a safeguards actuation (SI) test signal; and

c. By testing each cooler unit performance by measuring the temperature differential in the cooling water supply and air stream. The provisions of Specification 4.0.4 are not applicable for entry into Modes 3 and 4 for this specification.

d. By performing surveillance requirements of 4.6.2.2.a for the redundant coolers prior to initiating maintenance on the affected cooler unit being declared inoperable.

CONTAINMENT SYSTEMS

3/4.6.3 EMERGENCY CONTAINMENT FILTERING SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.3 Three emergency containment filtering units shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

With one emergency containment filtering unit inoperable, restore the inoperable filter to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.3 Each emergency containment filtering unit shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 15 minutes;

b. At least once per refueling or (1) after any structural maintenance on the HEPA filter or charcoal absorber housings, or (2) following operational exposure of filters to effluents from painting, fire or chemical releases, or (3) after every 720 hours of system operation by:

   1) Performance of a visual inspection for foreign material and gasket deterioration, and verifying that the filtering unit satisfies the in-place penetration and bypass leakage testing acceptance criteria of greater than or equal to 99% removal of DOP and halogenated hydrocarbon at the system flow rate of 37,500 cfm ± 10%;

   2) Verifying within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with applicable portion of C.6.b of Regulatory Guide 1.52, Revision 2, March 1978 and performed in accordance with ANSI N510-1975, meets the acceptance criteria of greater than 99.9% removal of elemental iodine; and that any charcoal failing to meet this criteria be replaced with charcoal that meets or exceeds the criteria of position C.6.a of Regulatory Guide 1.52, Rev. 2; and

3/4.6.3 EMERGENCY CONTAINMENT FILTERING SYSTEM (Continued)

3) Verifying a system flow rate of 37,500 cfm ± 10% and a pressure drop across the HEPA and charcoal filters of less than 6 inches water gauge during system operation.

c. After maintenance affecting flow distribution, by performance of a visual inspection and an air distribution test at a system flow rate of 37,500 cfm ± 10%;

d. At least once per refueling:

1) Verifying that the system starts on a Safety Injection test signal;

2) Verifying that the filter cooling spray solenoid valves can be opened by operator action and are opened automatically on a loss of flow signal.

e. After each complete or partial replacement of a HEPA filter bank, by performances of a visual inspection for foreign material and gasket deterioration, and verifying that the filtering unit satisfies the in-place penetration and bypass leakage testing acceptance criteria of greater than 99% removal of DOP aerosol while operating the system at a flow rate of 37,500 cfm ± 10%;

f. After each complete or partial replacement of a charcoal adsorber bank, by performances of a visual inspection for foreign material and gasket deterioration, and verifying that the filtering unit satisfies the in-place penetration and bypass leakage testing acceptance criteria of greater than or equal to 99% removal of halogenated hydrocarbon while operating the system at a flow rate of 37,500 cfm ± 10%; and

g. Prior to initiating maintenance on any inoperable system valve, by testing all valves that provide the duplicate function to demonstrate operability.

CONTAINMENT SYSTEMS

3/4.6.4 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.4 The containment isolation valves specified in Table 3.6-1 shall be OPERABLE with isolation times as shown in Table 3.6-1.

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

With one or more of the isolation valve(s) specified in Table 3.6-1 inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and:

a. Restore the inoperable valve(s) to OPERABLE status within 4 hours, or

b. Isolate each affected penetration within 4 hours by use of at least one deactivated automatic containment isolation valve secured in the isolation position, or

c. Isolate each affected penetration within 4 hours by use of at least one closed manual containment isolation valve or blind flange, or

d. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.1 The isolation valves specified in Table 3.6-1 shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by performance of a cycling test, and verification of isolation time.

4.6.4.2 Each isolation valve specified in Table 3.6-1 shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE at least once per 18 months by:

a. Verifying that on a Phase "A" Isolation test signal, each Phase "A" isolation valve actuates to its isolation position;

b. Verifying on a Phase "B" Isolation test signal, each Phase "B" isolation valve actuates to its isolation position; and

c. Verifying that on a Containment Ventilation Isolation test signal, each purge, exhaust, and instrument air bleed valve actuates to its isolation positon.

4.6.4.3 The isolation time of each power-operated or automatic valve of Table 3.6-1 shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

TABLE 3.6-1

CONTAINMENT ISOLATION VALVES

VALVE NUMBER		FUNCTION	MAXIMUM ISOLATION TIME (SECONDS)	NOTES
A. Phase "A" Isolation				
1.	CV-*-200A	Letdown Line	10	1,3
2.	CV-*-200B	Letdown Line	10	1,3
3.	CV-*-200C	Letdown Line	10	1,3
4.	CV-*-204	Letdown Line	10	1,3
5.	MOV-*-381	RCP Seal Water Leakoff Bypass	10	1,3,9
6.	CV-*-516	PRT Gas Analyzer Line	10	1,3
7.	CV-*-519A	PRT Makeup Primary Water Supply	60	1,3
8.	CV-*-855	N_2 Supply to Accumulators	10	1,3
9.	CV-*-956A	Pressurizer Steam Space Sample	10	1,3
10.	CV-*-956B	Pressurizer Liquid Space Sample	10	1,3
11.	CV-*-956D	Accumulator Sample Lines	10	1,3

1 Subject to Type C Test of 10CFR50, Appendix J
2 Opens on the ESF Signal
3 *-Applicable to Unit 3 and Unit 4
4 N/A - No specific closure time assumed in the LOCA accident analysis
5 May be opened on an intermittent basis under administrative control
6 N.A. - Not applicable
7 Isolation time requirement does not supercede requirement of IST program.
8 Subject to Type B test of 10CFR50, Appendix J
9 Isolation time may be exceeded provided it is less than 15 seconds and engineering is informed within a week.

TABLE 3.6-1
(continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (SECONDS)	NOTES
A. Phase "A" Isolation (Continued)			
12. CV-*-2821	Containment Sump Discharge	10	1,3
13. CV-*-2822	Containment Sump Discharge	10	1,3
14. SV-*-2911	Containment Air Sample	10	1,3
15. SV-*-2912	Containment Air Sample	10	1,3
16. SV-*-2913	Containment Air Sample	10	1,3
17. CV-*-4658A	RC Drain Tank Vent	10	1,3
18. CV-*-4658B	RC Drain Tank Vent	10	1,3
19. CV-*-4659A	RC Drain Tank Line to H_2 Analyzer	10	1,3
20. CV-*-4659B	RC Drain Tank Line to H_2 Analyzer	10.	1,3
21. CV-*-4668A	RC Drain Tank Pump Discharge	10	1,3
22. CV-*-4668B	RC Drain Tank Pump Discharge	10	1,3
23. CV-*-6165	Breathing Air	10	1,3
24. SV-*-6385	PRT Gas Analyzer Line	10	1,3
25. MOV-*-6386	Excess Letdown and RCP Seal Water Return to CVCS	18	1,3
26. SV-*-6428	RCS Sample	10	1,3

1 Subject to Type C Test of 10CFR50, Appendix J
2 Opens on the ESF Signal
3 *-Applicable to Unit 3 and Unit 4
4 N/A - No specific closure time assumed in the LOCA accident analysis
5 May be opened on an intermittent basis under administrative control
6 N.A. - Not applicable
7 Isolation time requirement does not supercede requirement of IST program.
8 Subject to Type B test of 10CFR50, Appendix J
9 Isolation time may be exceeded provided it is less than 15 seconds and engineering is informed within a week.

TABLE 3.6-1
(continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER		FUNCTION	MAXIMUM ISOLATION TIME (SECONDS)	NOTES
B. Phase "B" Isolation				
1.	FCV-*-626	CC Return from RCP	N/A	3,7,4
2.	MOV-*-716A	CC Supply to RCP	N/A	3,7,4
3.	MOV-*-716B	CC Supply to RCP	N/A	3,7,4
4.	MOV-*-730	CC Return from RCP Thermal Barrier Coolers	N/A	3,7,4
C. Containment Ventilation Isolation				
1.	POV-*-2600	Containment Purge Supply	5	1,3
2.	POV-*-2601	Containment Purge Supply	5	1,3
3.	POV-*-2602	Containment Purge Exhaust	5	1,3
4.	POV-*-2603	Containment Purge Exhaust	5	1,3
5.	CV-*-2819	Instrument Air Bleed	10	1,3
6.	CV-*-2826	Instrument Air Bleed	10	1,3
D. Manual				
1.	*-333	Charging Line	N.A.	1,3,5,6
2.	HV-*-1	Post Accident Containment Vent System	N.A.	1,3,5,6
3.	HV-*-2	Post Accident Containment Vent System	N.A.	1,3,5,6
4.	*-895V	SI Test and Purge Line	N.A.	1,3,5,6

1 Subject to Type C Test of 10CFR50, Appendix J
2 Opens on the ESF Signal
3 *-Applicable to Unit 3 and Unit 4
4 N/A - No specific closure time assumed in the LOCA accident analysis
5 May be opened on an intermittent basis under administrative control
6 N.A. - Not applicable
7 Isolation time requirement does not supercede requirement of IST program.
8 Subject to Type B test of 10CFR50, Appendix J
9 Isolation time may be exceeded provided it is less than 15 seconds and engineering is informed within a week.

TABLE 3.6-1
(continued)

CONTAINMENT ISOLATION VALVES

	VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (SECONDS)	NOTES
D. Manual				
5.	*-40-204	Service Air	N.A.	1,3,5,6
6.	*10-582	Demineralized Water Supply	N.A.	1,3,5,6
7.	Hv-4-3	U-4 H_2 Monitoring	N.A.	1,5,6
8.	Hv-4-4	U-4 H_2 Monitoring	N.A.	1,5,6
9.	Hv-3-3	U-3 H_2 Monitoring	N.A.	1,5,6
10.	Hv-3-4	U-3 H_2 Monitoring	N.A.	1,5,6
11.	*-2025	ILLRT Test Connection	N.A.	1,3,5,6
12.	*-2026	ILLRT Test Connection	N.A.	1,3,5,6
13.	PAHM-*-002A	Containment Vent to Hydrogen Analyzer A	N.A.	1,3,5,6
14.	PAHM-*-002B	Containment Vent to Hydrogen Analyzer B	N.A.	1,3,5,6
15.	PAHM-*-001A	Hydrogen Analyzer A Return	N.A.	1,3,5,6
16.	PAHM-*-001B	Hydrogen Analyzer B Return	N.A.	1,3,5,6
17.	HV-*-17	Post Accident Containment Vent	N.A.	1,3,5,6
18.	*-2023	PZR Deadweight Tester	N.A.	1,3,5,6
19.	*-2024	PZR Deadweight Tester	N.A.	1,3,5,6
20.	PCV-*-1014	Nitrogen Syupply to RCDT	N.A.	1,3,5,6
E. Remote Manual				
1.	CV-*-519B	PRT M.U. P.W. Supply	N.A.	1,3,5,6

1 Subject to Type C Test of 10CFR50, Appendix J
2 Opens on the ESF Signal
3 *-Applicable to Unit 3 and Unit 4
4 N/A - No specific closure time assumed in the LOCA accident analysis
5 May be opened on an intermittent basis under administrative control
6 N.A. - Not applicable
7 Isolation time requirement does not supercede requirement of IST program.
8 Subject to Type B test of 10CFR50, Appendix J
9 Isolation time may be exceeded provided it is less than 15 seconds and engineering is informed within a week.

TABLE 3.6-1
(continued)

CONTAINMENT ISOLATION VALVES

	VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (SECONDS)	NOTES
E. Remote Manual (Continued)				
2.	CV-*-522A	PW Supply to RCP A	N.A.	1,3,5,6
3.	CV-*-522B	PW Supply to RCP B	N.A.	1,3,5,6
4.	CV-*-522C	PW Supply to RCP C	N.A.	1,3,5,6
5.	CV-*-951	Pressurizer Steam Space Sample	N.A.	1,3,5,6
6.	CV-*-953	Pressurizer Liquid Space Sample	N.A.	1,3,5,6
7.	SV-*-6427A	Hot Leg RCS Sample	N.A.	1,3,5,6,7
8.	SV-*-6427B	Hot Leg RCS Sample	N.A.	1,3,5,6,7
9.	MOV-*-860A	SI Recirculation from Containment Sump	N.A.	1,3,5,6,7
10.	MOV-*-861A	SI Recirculation from Containment Sump	N.A.	1,3,5,6,7
11.	MOV-*-860B	SI Recirculation from Containment Sump	N.A.	1,3,5,6,7
12.	MOV-*-861B	SI Recirculation from Containment Sump	N.A.	1,3,5,6,7

1 Subject to Type C Test of 10CFR50, Appendix J
2 Opens on the ESF Signal
3 *-Applicable to Unit 3 and Unit 4
4 N/A - No specific closure time assumed in the LOCA accident analysis
5 May be opened on an intermittent basis under administrative control
6 N.A. - Not applicable
7 Isolation time requirement does not supercede requirement of IST program.
8 Subject to Type B test of 10CFR50, Appendix J
9 Isolation time may be exceeded provided it is less than 15 seconds and engineering is informed within a week.

TABLE 3.6-1
(Continued)

CONTAINMENT ISOLATION VALVES

	VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (SECONDS)	NOTES
E.	**Remote Manual (Continued)**			
13.	CV-*-955C	Accumulator Sample Lines	N.A.	1,3,5,6
14.	CV-*-955D	Accumulator Sample Lines	N.A.	1,3,5,6
15.	CV-*-955E	Accumulator Sample Lines	N.A.	1,3,5,6
16.	HCV-*-121	Charging Line	N.A.	1,3,5,6
17.	MOV-*-872	Alternate Low Head SI to RCS	N.A.	1,3,5,6,7
18.	MOV-*-880A	Containment Spray Pump Discharge	N/A	1,2,3,7
19.	MOV-*-880B	Containment Spray Pump Discharge	N/A	1,2,3,7
F.	**Check Valves**			
1.	*-519	PRT Nitrogen Supply	N.A.	1,3,5,6
2.	*-518	PRT Nitrogen Supply	N.A.	1,3,5,6
3.	*-312C	Charging Line	N.A.	1,3,5,6
4.	*-890A	Containment Spray Line	N.A.	1,3,5,6
5.	*-890B	Containment Spray Line	N.A.	1,3,5,6
6.	*-298A	Seal Injection Water to RCP A	N.A.	1,3,5,6
7.	*-298B	Seal Injection Water to RCP B	N.A.	1,3,5,6

1 Subject to Type C Test of 10CFR50, Appendix J
2 Opens on the ESF Signal
3 *-Applicable to Unit 3 and Unit 4
4 N/A - No specific closure time assumed in the LOCA accident analysis
5 May be opened on an intermittent basis under administrative control
6 N.A. - Not applicable
7 Isolation time requirement does not supercede requirement of IST program.
8 Subject to Type B test of 10CFR50, Appendix J
9 Isolation time may be exceeded provided it is less than 15 seconds and engineering is informed within a week.

TABLE 3.6-1
(Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER		FUNCTION	MAXIMUM ISOLATION TIME (SECONDS)	NOTES
F. Check Valves (Continued)				
8.	*-298C	Seal Injection Water to RCP C	N.A.	1,3,5,6
9.	*-40-340A	Instrument Air Supply	N.A.	1,3,5,6
10.	*-40-336	Instrument Air Supply	N.A.	1,3,5,6
11.	*-BA-201	Breathing Air	N.A.	1,3,5,6
12.	*-11-003	Containment Air Sample	N.A.	1,3,5,6
13.	*-40-205	Service Air	N.A.	1,3,5,6
14.	*-945E	N_2 Supply to Accumulators	N.A.	1,3,5,6
15.	*-10-567	Demineralized Water Supply	N.A.	1,3,5,6
G. Flanges				
1.	Fuel Transfer Tube		N.A.	3,6,8
2.	Containment Test Penetration 65A		N.A.	1,3,6,8
3.	Containment Test Penetration 65B		N.A.	1,3,6,8
4.	Containment Test Penetration 65C		N.A.	1,3,6,8

1 Subject to Type C Test of 10CFR50, Appendix J
2 Opens on the ESF Signal
3 *-Applicable to Unit 3 and Unit 4
4 N/A - No specific closure time assumed in the LOCA accident analysis
5 May be opened on an intermittent basis under administrative control
6 N.A. - Not applicable
7 Isolation time requirement does not supercede requirement of IST program.
8 Subject to Type B test of 10CFR50, Appendix J
9 Isolation time may be exceeded provided it is less than 15 seconds and engineering is informed within a week.

3/4.6.5 COMBUSTIBLE GAS CONTROL MONITORS

LIMITING CONDITION FOR OPERATION

3.6.5 Two independent containment hydrogen monitors shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With one hydrogen monitor inoperable, restore the inoperable monitor to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours.

b. With both hydrogen monitors inoperable, restore at least one monitor to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.1 Each hydrogen monitor shall be demonstrated OPERABLE by the performance of a CHANNEL CHECK at least once per 12 hours, an ANALOG CHANNEL OPERATIONAL TEST at least once per 31 days, and at least once per 92 days on a STAGGERED TEST BASIS by performing a CHANNEL CALIBRATION using sample gas containing:

a. One volume percent hydrogen, balance nitrogen, and

b. Four volume percent hydrogen, balance nitrogen.

4.6.5.2 The flow path to each hydrogen monitor shall be demonstrated OPERABLE at least once per 31 days by a system walkdown to verify that each accessible manual, power operated, or automatic valve is in its correct position and that power is available to those components related to the operability of the flow path.

3/4.6.6 POST ACCIDENT CONTAINMENT VENT SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.6 A Post Accident Containment Vent System shall be OPERABLE.

APPLICABILITY: MODES 1 and 2

ACTION:

With the Post Accident Containment Vent System inoperable, restore the Post Accident Containment Vent System to OPERABLE status within 7 days or be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.6 The Post Accident Containment Vent System shall be demonstrated OPERABLE:

a. At least once per 31 days, by demonstrating system flow path operability via a system walkdown to verify that each accessible manual valve is in its correct position.

b. At least once per 18 months or (1) after any structural maintenance of the HEPA filter or charcoal adsorber housings, or (2) following operational exposure of filters to effluents from painting, fire or chemical release in any ventilation zone communicating with the system, or (3) after 720 hours of system operation, or (4) after replacement of a filter by:

1) A visual inspection of the system for foreign materials and gasket deterioration and verifying that the filters system satisfys the in-place penetration and bypass leakage testing acceptance criteria of less than 1% for DOP and halogenated hydrocarbon tests conducted at the design flow rate of 55 cfm ± 10%;

2) Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample performed in accordance with ANSI N510-1975, meets the methyl iodide removal criteria of greater than or equal to 90%, and that any charcoal failing to meet the criteria be replaced with charcoal that meets or exceeds the criteria of Position C.6.a of Regulatory Guide 1.52, Revision 2.

SURVEILLANCE REQUIREMENTS

c. Prior to initiating maintenance on any inoperable valve, by testing all valves that provide the duplicate function to verify their operability.

d. At least once per 18 months by:

1) Verifying that the pressure drop across the combined HEPA filter and charcoal absorber is less than 6 inches water gauge at a flow rate of 55 cfm ± 10%;.

2) Visual inspection of the system and operation of all valves.

3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam line Code safety valves associated with each steam generator shall be OPERABLE with lift settings as specified in Table 3.7-1.

APPLICABILITY: MODES 1, 2 and 3

ACTION:

a. With 3 reactor coolant loops and associated steam generators in operation and with one or more main steam line Code safety valves inoperable, operation in MODES 1, 2 and 3 may proceed provided, that within 4 hours, either the inoperable valve is restored to OPERABLE status or the Power Range Neutron Flux High Trip Setpoint is reduced per Table 3.7-2; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.1 Every refueling shutdown, the lift settings of the main steam line code safety valves shall be verified to be within the limits specified in Table 3.7-1.

TABLE 3.7-1

STEAM LINE SAFETY VALVES PER LOOP

VALVE NUMBER Loop A, Loop B, Loop C	LIFT SETTING (± 1%)*	ORIFICE SIZE SQUARE INCHES
1. RV1400, RV1405, RV1410	1085 psig	16
2. RV1401, RV1406, RV1411	1100 psig	16
3. RV1402, RV1407, RV1412	1115 psig	16
4. RV1403, RV1408, RV1413	1130 psig	16

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

TABLE 3.7-2

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES

MAXIMUM NUMBER OF INOPERABLE SAFETY VALVES ON ANY STEAM GENERATOR	MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT (PERCENT OF RATED THERMAL POWER)
1	82
2	54
3	27

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 Two independent auxiliary feedwater trains as specified in Table 3.7-3 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3

ACTION:

1) With one of the two required independent auxiliary feedwater trains inoperable, either restore the inoperable train to an OPERABLE status within 72 hours, or place the affected unit(s) in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

2) With both required auxiliary feedwater trains inoperable, within 1 hour either restore both trains to an OPERABLE status, or restore one train to an OPERABLE status and follow ACTION statement 1 above for the other train, or place the affected unit(s) in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.2.1 The required independent auxiliary feedwater trains shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by:

   1) Verifying by control panel indication and visual observation of equipment that each steam turbine-driven pump operates for 15 minutes or greater and develops a flow of greater than or equal to 373 gpm to the entrance of the steam generators. The provisions of Specification 4.0.4 are not applicable for entry into MODES 2 and 3;

   2) Verifying by control panel indication and visual observation of equipment that the auxiliary feedwater discharge valves and the steam supply and turbine pressure valves operate as required to deliver the required flow during the pump performance test above;

   3) Verifying that each non-automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in its correct position; and

SURVEILLANCE REQUIREMENTS (Continued)

4) Verifying that power is available to those components which require power for flow path operability.

b. At least once per 18 months by:

1) Verifying that each automatic valve in the flow path actuates to its correct position upon receipt of each Auxiliary Feedwater Actuation test signal, and

2) Verifying that each auxiliary feedwater pump receives a start signal as designed automatically upon receipt of each Auxiliary Feedwater Actuation test signal.

4.7.1.2.2 An auxiliary feedwater flow path to each steam generator shall be demonstrated OPERABLE following each COLD SHUTDOWN of greater than 30 days prior to entering MODE 1 by verifying normal flow to each steam generator.

TABLE 3.7-3

AUXILIARY FEEDWATER SYSTEM OPERABILITY

Unit	Train	Steam Supply Flowpath(4)	Pump	Discharge Water Flowpath(3 and 4)
3	1	SG 3C via MOV-3-1405 or SG 3B via MOV-3-1404(1)	A or C(2)	SG 3A via CV-3-2816 SG 3B via CV-3-2817 SG 3C via CV-3-2818
3	2	SG 3A via MOV-3-1403 or SG 3B via MOV-3-1404(1)	B or C(2)	SG 3A via CV-3-2831 SG 3B via CV-3-2832 SG 3C via CV-3-2833
4	1	SG 4C via MOV-4-1405 or SG 4B via MOV-4-1404(1)	A or C(2)	SG 4A via CV-4-2816 SG 4B via CV-4-2817 SG 4C via CV-4-2818
4	2	SG 4A via MOV-4-1403 or SG 4B via MOV-4-1404(1)	B or C(2)	SG 4A via CV-4-2831 SG 4B via CV-4-2832 SG 4C via CV-4-2833

NOTES

(1) Steam admission valves MOV-3-1404 and MOV-4-1404 can be aligned to either train to restore operability in the event MOV-3-1403 or MOV-3-1405, or MOV-4-1403 or MOV-4-1405 are inoperable.

(2) During single and two unit operation, one pump is required to be operable in each train. The standby pump "C" can be aligned to either train to restore operability in the event one of the required pumps is inoperable.

(3) One flow control valve in each train for each unit can be inoperable for a period not to exceed 72 hours. The ACTION required for a single train inoperable shall be followed.

(4) If any local manual realignment of valves is required when operating the Auxiliary Feedwater pumps, a dedicated indidivual, who is in communication with the control room, shall be stationed at the auxiliary pump area. Upon instructions from the control room, this operator would realign the valves in the AFW system train to its normal operational alignment.

PLANT SYSTEMS

CONDENSATE STORAGE TANKS

LIMITING CONDITION FOR OPERATION

3.7.1.3 The Condensate Storage Tanks shall be OPERABLE with a contained water volume of at least 185,000 gallons of water as follows:

Single Unit Prior to Escalating into Mode 3

a) ONE water supply from either Condensate Storage Tank including flowpath piping and valves.

Second Unit Prior to Escalating into Mode 3

a) ONE water supply from each unit's corresponding Condensate Storage Tank including flowpath piping and valves.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

Single Unit at or Above Mode 3

1) With one water supply from a Condensate Storage Tank inoperable, within 4 hours, either realign the other Condensate Storage Tank containing the required water volume to the suction of the Auxiliary Feedwater pumps or restore the inoperable water supply to OPERABLE status or be in HOT STANDBY in the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

2) With both water supplies from the Condensate Storage Tanks inoperable, within 4 hours restore the water supply from either Condensate Storage Tank to OPERABLE status or be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

Both Units at or Above Mode 3

1) With one water supply from a Condensate Storage Tank inoperable, restore the inoperable water supply to OPERABLE status within 4 hours or place one unit in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. Refer to Single Unit Operation ACTION for single unit at or above MODE 3.

LIMITING CONDITION FOR OPERATION (CONTINUED)

ACTION (Continued):

2) With both water supplies from the Condensate Storage Tank inoperable within 1 hour restore one water supply from a Condensate Storage Tank to OPERABLE status or place one unit in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. If unable to restore at least one water supply from a Condensate Storage Tank to OPERABLE status within 4 hours from initial declaration of inoperability, the second unit shall be placed in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.3 The Condensate Storage Tanks shall be demonstrated OPERABLE at least once per 12 hours by verifying the contained water volume is within its limit when the tank is the supply source for the auxiliary feedwater pumps.

PLANT SYSTEMS

SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.7.1.4 The specific activity of the Secondary Coolant System shall be less than or equal to 0.10 microCuries/gram DOSE EQUIVALENT I-131.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the specific activity of the secondary coolant system greater than 0.10 microCuries/gram DOSE EQUIVALENT I-131, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.4 The specific activity of the secondary coolant system shall be determined to be within the limit by performance of the sampling and analysis program of Table 4.7-1.

TABLE 4.7-1

SECONDARY COOLANT SYSTEM SPECIFIC ACTIVITY

SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY
1. Gross Radioactivity Determination	3 Times per 7 Days With A Maximum Time of 72 Hours Between Samples
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	a) 1 per 31 days, whenever the gross activity determination indicates iodine concentrations greater than 10% of the allowable limit. b) 1 per 6 months, whenever the gross activity determination indicates iodine concentrations below 10% of the allowable limit.

PLANT SYSTEMS

MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam line isolation valve (MSIV) shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

MODE 1:

With one MSIV inoperable but open, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 4 hours; otherwise, reduce power to less than or equal to 5% of RATED THERMAL POWER within the next 6 hours.

MODES 2 and 3:

With one MSIV inoperable, subsequent operation in MODE 2 and 3 may proceed provided the isolation valve is maintained closed. Otherwise, be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each MSIV shall be demonstrated OPERABLE by verifying full closure within 5 seconds when tested pursuant to Specification 4.0.5. The provision of Specification 4.0.4 are not applicable for entry into MODES 2 and 3.

3/4.7.1.6 STANDBY FEEDWATER SYSTEM

3.7.1.6 Two standby feedwater pumps shall be available* and at least 60,000 gallons of water (available volume), shall be in the Demineralized Water Storage Tank **.

APPLICABILITY: MODES 1, 2, 3

ACTION:

a. With one standby feedwater pump unavailable, restore the unavailable pump to available status within 30 days or submit a SPECIAL REPORT per 3.7.1.6.d.

b. With both standby feedwater pumps unavailable:

   1. Within 24 hours, notify the NRC and provide cause for unavailability and plans to restore pump(s) to available status and,

   2. Submit a SPECIAL REPORT per 3.7.1.6.d.

c. With less than 60,000 gallons of water in the Demineralized Water Storage Tank restore the available volume to at least 60,000 gallons within 24 hours or submit a SPECIAL REPORT per 3.7.1.6.d.

d. If a SPECIAL REPORT is required per the above specifications submit a report describing the cause of the unavailability, action taken and a schedule for restoration within 30 days in accordance with 6.9.2

e. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.1 The Demineralized Water Storage tank water volume shall be determined to be within limits at least once per 24 hours.

4.7.1.2 At least monthly verify the standby feedwater pumps are available by testing in recirculation on a STAGGERED TEST BASIS.

4.7.1.3 During each refueling outage, verify availability of the respective standby feedwater pump by powering from the non-safety grade diesel generators and providing feedwater to the steam generators.

* These pumps are not safety related equipment and do not require plant safety related emergency power sources for availability.

** The Demineralized Water Storage Tank is non-safety grade.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

LIMITING CONDITION FOR OPERATION

3.7.2 The temperatures of both the reactor and secondary coolants in the steam generators shall be greater than 70 F when the pressure of either coolant in the steam generator is greater than 200 psig.

APPLICABILITY: At all times.

ACTION:

With the requirements of the above specification not satisfied:

a. Reduce the steam generator pressure of the applicable side to less than or equal to 200 psig within 30 minutes, and

b. Perform an engineering evaluation to determine the effect of the overpressurization on the structural integrity of the steam generator. Determine that the steam generator remains acceptable for continued operation prior to increasing its temperatures above 200 F.

SURVEILLANCE REQUIREMENTS

4.7.2 The pressure in each side of the steam generator shall be determined to be less than 200 psig at least once per hour when the temperature of either the reactor or secondary coolant is less than 70 F.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3 The Component Cooling Water Sytem shall be OPERABLE as follows:

a. Two OPERABLE component cooling water pumps with independent power supplies, and

b. Three OPERABLE component cooling water heat exchangers, and

c. Two OPERABLE component cooling water headers.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

1. With only one component cooling water pump OPERABLE, restore at least two component cooling water pumps to OPERABLE status in 72 hours or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

2. With only two component cooling heat exchangers OPERABLE, restore the inoperable heat exchanger to OPERABLE status in 72 hours or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

3. With only one component cooling header OPERABLE, restore the inoperable header to OPERABLE status in 72 hours or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.3 The Component Cooling Water System shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) servicing safety-related equipment that is not locked, sealed or otherwise secured in position is in its correct position and that power is available to flow path components which require power for operability; and

b. At least once per refueling by verifying that:

1) Each automatic valve servicing safety-related equipment actuates to its correct position on an SI test signal, and

2) The Component Cooling Water System pumps required by 3.7.3 start automatically on an SI test signal or as part of the sequencer action following a power supply undervoltage signal.

3/4.7.4 INTAKE COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.4 At least two intake cooling water headers and two intake cooling water pumps with independent power supplies shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

With only one intake cooling water header or pump OPERABLE, restore at least two headers and two pumps with independent power supplies to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.4 At least two intake cooling water loops shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) servicing safety-related equipment that is not locked, sealed or otherwise secured in position is in its correct position and that power is available to flow path components which require power for operation, and

b. At least once per refueling by verifying that:

 1) Each automatic valve servicing safety-related equipment actuates to its correct position on an SI test signal, and

 2) Each intake cooling water pump starts automatically on a safety injection or bus under-voltage test signal.

PLANT SYSTEM

3/4.7.5 ULTIMATE HEAT SINK

LIMITING CONDITION FOR OPERATION

3.7.5 The ultimate heat sink shall be OPERABLE with an average supply water temperature to the Intake Cooling Water System less than or equal to 95 F.

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

With Ultimate Heat Sink average supply water temperature greater than 95 F, within 72 hours perform an engineering evaluation to confirm the capability of the Intake Cooling Water System to meet the system design basis for the accident case, or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.5.1 The Ultimate Heat Sink shall be determined OPERABLE at least once per 24 hours by verifying that the average supply water temperature is less than or equal to 95 F.

4.7.5.2 The Component Cooling Water heat exchangers shall be tested on a STAGGERED TEST BASIS once per 92 days to determine the heat transfer coefficient.

3/4.7.6 FLOOD PROTECTION

LIMITING CONDITION FOR OPERATION

3.7.6 Flood protection shall be provided as follows for the facility:

a. With either a Hurricane Watch or a Hurricane Warning issued for the facility, stoplogs shall be removed from storage and prepared for installation.

b. With a Hurricane Warning issued for the facility, the stoplogs shall be installed on the plant flood protection wall at locations specified in Table 3.7-4.

APPLICABILITY: At all times.

ACTION:

a. With the above flood protection requirements not implemented, take immediate corrective action to provide flood protection.

b. If the physical conditions of the stoplogs are found to be unacceptable, take corrective action to replace or repair the affected stoplogs.

SURVEILLANCE REQUIREMENTS

4.7.6.1 At least once per 12 hours while the stoplogs are required to be in place, verify the stoplogs are properly installed provided personnel safety is not compromised.

4.7.6.2 At least once per 18 months, perform a visual inspection of the physical conditions of the stoplogs to verify operability.

4.7.6.3 Meteorological forecasts shall be obtained from the National Hurricane Center in Miami, Florida at least once per 6 hours during either a Hurricane Watch or a Hurricane Warning. NOAA may be used as a alternate source for Meteorological forecasts.

TABLE 3.7-4

FLOOD PROTECTION STOPLOG LOCATIONS

1. By the Unit 3 4160 Volt Switchgear Room entrance.
2. By the Diesel Oil Storage Tank Dike Area.
3. By the Unit 3 and Unit 4 Main Transformers.
4. By the Unit 4 Steam Generator Feed Pump Room.
5. By the Unit 4 Blowdown Tank.
6. On the entrance to the Unit 3 CCW Pump Area.
7. On the entrance to the Unit 4 CCW Pump Area.
8. By the Unit 3 and Unit 4 New Fuel Storage Area.
9. By the Unit 3 and Unit 4 Lube Oil Reservoir.
10. On the entrance to the Unit 3 and Unit 4 Condenser Pits.
11. On the entrance to the Unit 3 and Unit 4 Spent Fuel Pit Heat Exchanger Rooms.
12. On the entrance to the Auxiliary Building Chemical Storage Area.

3/4.7.7 CONTROL ROOM EMERGENCY AIR CLEANUP

LIMITING CONDITION FOR OPERATION

3.7.7 The Control Room Ventilation System shall be OPERABLE.

APPLICABILITY: ALL MODES.

ACTION:

MODES 1, 2, 3, and 4:

With the Control Room Ventilation System inoperable, restore the inoperable system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

MODES 5 and 6:

With the Control Room Ventilation System inoperable, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.7.7 The Control Room Ventilation System shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the control room air temperature is less than or equal to 120 F;

b. At least once per 31 days by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 15 minutes;

c. At least once per 18 months or (1) after every 720 hours of system operation (2) after any structural maintenance on the HEPA filter or charcoal adsorber housings or filter replacement, or (3) following operational exposure of the filters to effluents from painting, fire, or chemical release in any ventilation zone communicating with the system by:

   1) Performing a visual inspection for any foreign materials and gasket deterioration in the HEPA filters and charcoal adsorbers.

2) Verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of greater than or equal to 99% DOP and halogenated hydrocarbon removal and the system flow rate is 1000 cfm ± 10%;

3) Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978 and analyzed per ANSI N510-1975, meets the criteria for methyl iodide removal efficiency of greater than or equal to 90%, or the charcoal be replaced with charcoal that meets or exceeds the criteria of position C.6.a of Regulatory Guide 1.52 (Revision 2); and

d. At least once per 12 months by verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches Water Gauge at the design flow rate of 1000 cfm ± 10%.

e. At least once per 18 months, by verifying that on a Containment Phase "A" Isolation test signal, the system automatically switches into a recirculation mode of operation.

f. Before initiating maintenance on any inoperable system dampers, by verifying that all dampers that provide duplicate function are OPERABLE.

PLANT SYSTEMS

3/4.7.8 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.8 All safety related snubbers shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4. MODES 5 and 6 for snubbers located on systems required OPERABLE in those MODES.

ACTION:

With one or more snubbers inoperable within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.7.8f. on the component to which the snubber was attached or declare the attached system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.8 Each safety related snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program in lieu of the requirements of Specification 4.0.5.

a. Inspection Types

As used in this specification, type of snubber shall mean snubbers of the same design and manufacturer, irrespective of capacity.

b. Visual Inspections

Snubbers may be categorized as inaccessible or accessible during reactor operation. Each of these groups (inaccessible and accessible) may be inspected independently according to the schedule below. The first inservice visual inspection of each type of snubber shall be performed after 4 months but within 10 months of commencing POWER OPERATION and shall include all snubbers. If all snubbers of each type (on any system) are found OPERABLE during the first inservice visual

b. Visual Inspections (Continued)

inspection, the second inservice visual inspection (of that system) shall be performed at the first refueling outage. Otherwise, subsequent visual inspections (of a given system) shall be performed in accordance with the following schedule:

No. of Inoperable Snubbers of Each Type (on any system) per Inspection Period per Unit	Subsequent Visual Inspection Period* **
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3, 4	124 days ± 25%
5, 6, 7	62 days ± 25%
8 or more	31 days ± 25%

c. Visual Inspection Acceptance Criteria

Visual inspections shall verify that: (1) there are no visible indications of damage or impaired OPERABILITY, (2) attachments to the foundation or supporting structure are secure, and (3) fasteners for attachment of the snubber to the component and to the snubber anchorage are secure. Snubbers which appear inoperable as a result of visual inspections may be determined OPERABLE for the purpose of establishing the next visual inspection interval, provided that: (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers that may be generically susceptible; and (2) the affected snubber is functionally tested in the as-found condition and determined OPERABLE per Specification 4.7.8e.

* The inspection interval for each type of snubber shall not be lengthened more than one step at a time unless a generic problem has been identified and corrected; in that event the inspection interval may be lengthened one step the first time and two steps thereafter if no inoperable snubbers of that type are found.

** The provisions of Specification 4.0.2 are not applicable.

d. Functional Tests

For each unit during refueling shutdown, a representative sample of snubbers shall be tested using the following sample plan:

1) At least 10% of the total number of safety ralated snubbers for the respective unit identified by site records shall be functionally tested either in-place or in a bench test. For each snubber of a type that does not meet the functional test acceptance criteria of Specification 4.7.8e, an additional 10% of that type of snubber shall be functionally tested until no more failures are found or until all snubbers of that type have been functionally tested;

2) The representative sample selected for functional testing shall include the various configurations, operating environments and the range of size and capacity of snubbers. At least 25% of the snubbers in the representative sample shall include snubbers from the following categories;

   A. Snubbers within 5 feet of heavy equipment (ex. valves, pumps, turbines, motors, etc.)

   B. Snubbers within 10 feet of the discharge from a safety relief valve.

3) Snubbers identified by site records as "Especially Difficult to Remove" or in "High Radiation Zones During Shutdown" shall also be included in the representative sample.*

   In addition to the regular sample, snubbers which failed the previous functional test shall be retested during the next test period. If a spare snubber has been installed in place of a failed snubber, then both the failed snubber (if it is repaired and installed in another position) and the spare snubber shall be retested. Test results of these snubbers may not be included for the re-sampling.

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* Permanent or other exemptions from functional testing for individual snubbers in these categories may be granted by the Commission only if a justifiable basis for exemption is presented and/or snubber life destructive testing was performed to qualify snubber OPERABILITY for all design conditions at either the completion of their fabrication or at a subsequent date.

SURVEILLANCE REQUIREMENTS (Continued)

e. Mechanical Snubbers Functional Test Acceptance Criteria

The snubber functional test shall verify that:

1) Activation (restraining action) is achieved within the specified range of velocity or acceleration in both tension and compression;

2) Snubber release rate, where required, is within the specified range in tension and compression,

3) The force required to initiate or maintain motion of the snubber is within the specified range in both directions of travel.

f. Functional Test Failure Analysis

An engineering evaluation shall be made of each failure to meet the functional test acceptance criteria to determine the cause of the failure. The results of this evaluation shall be used, if applicable, in selecting snubbers to be tested in an effort to determine the OPERABILITY of other snubbers irrespective of type which may be subject to the same failure mode.

If any snubber selected for functional testing either fails to activate or fails to move, i.e., frozen in place, the cause will be evaluated under the provisions of 10 CFR Part 21.

Should the results of the evaluation indicate that the failure was caused by either manufacturer or design deficiency, further action shall be taken, if needed, based on manufacturer or engineering recommendations.

For the snubber(s) found inoperable, an evaluation shall be performed on the components to which the inoperable snubber(s) are attached. The purpose of this evaluation shall be to determine if the components to which the inoperable snubber(s) are attached were adversely affected by the inoperability of the snubber(s) in order to ensure that the component remains capable of meeting the designed service.

SURVEILLANCE REQUIREMENTS (Continued)

g. Snubber Service Life Monitoring Program

A record of the service life of each snubber, the date at which the designated service life commences and the installation and maintenance records on which the designated service life is based shall be maintained as required by Specification 6.10.3m.

Concurrent with the first inservice visual inspection and during refueling shutdown thereafter, the installation and maintenance records for each safety related snubber as identified by site records shall be reviewed to verify that the indicated service life has not been exceeded or will not be exceeded prior to the next scheduled snubber service life review, the snubber service life shall be re-evaluated or the snubber shall be replaced or reconditioned so as to extend its service life beyond the date of the next scheduled service life review. This re-evaluation, replacement or reconditioning shall be indicated in the records.

3/4.7.9 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

3.7.9 Each sealed source containing radioactive material either in excess of 100 microCuries of beta and/or gamma emitting material or 5 microCuries of alpha emitting material shall be free of greater than or equal to 0.005 microCurie of removable contamination.

APPLICABILITY: At all times.

ACTION:

a. With a sealed source having removable contamination in excess of the above limits, immediately withdraw the sealed source from use and either:

    1. Decontaminate and repair the sealed source, or

    2. Dispose of the sealed source in accordance with Commission Regulations.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.9.1 Test Requirements - Each sealed source shall be tested for leakage and/or contamination by:

a. The licensee, or

b. Other persons specifically authorized by the Commission or an Agreement State.

The test method shall have a detection sensitivity of at least 0.005 microCurie per test sample.

4.7.9.2 Test Frequencies - Each category of sealed sources (excluding startup sources and fission detectors previously subjected to core flux) shall be tested at the frequency described below:

SURVEILLANCE REQUIREMENTS (Continued)

a. Sources in use - At least once per 6 months for all sealed sources containing radioactive materials:

   1) with a half-life greater than 30 days (excluding Hydrogen 3), and

   2) In any form other than gas.

b. Stored sources not in use - Each sealed source and fission detector shall be tested prior to use or transfer to another licensee unless tested within the previous 6 months. Sealed sources and fission detectors transferred without a certificate indicating the last test date shall be tested prior to being placed into use; and

c. Startup sources and fission detectors - Each sealed startup source and fission detector shall be tested within 31 days prior to being subjected to core flux or installed in the core and following repair or maintenance to the source.

4.7.9.3 Reports - A report shall be prepared and submitted to the Commission on an annual basis if sealed source or fission detector leakage tests reveal the presence of greater than or equal to 0.005 μCurie of removable contamination.

4.7.9.4 A complete inventory of licensed radioactive materials in possession shall be maintained current at all times.

3/4.8 ELECTRICAL POWER SYSTEMS TECHNICAL SPECIFICATIONS

3/4.8.1.1	A.C. SOURCES OPERATING	LATER
3/4.8.1.2	A.C. SOURCES SHUTDOWN	LATER
3/4.8.2.1	D.C. SOURCES OPERATING	LATER
3/4.8.2.2	D.C. SOURCES SHUTDOWN	LATER
3/4.8.3.1	ONSITE POWER DISTRIBUTION OPERATING	LATER
3/4.8.3.2	ONSITE POWER DISTRIBUTION SHUTDOWN	LATER

3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 The boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of the following reactivity conditions is met; either:

a. A K_{eff} of 0.90 or less or

b. A boron concentration of greater than or equal to 1950 ppm.

APPLICABILITY: MODE 6.*

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate and continue boration at greater than or equal to 10 gpm of a solution containing greater than or equal to 20,000 ppm boron or its equivalent until K_{eff} is reduced to less than or equal to 0.90 or the boron concentration is restored to greater than or equal to 1950 ppm, whichever is the more restrictive.

SURVEILLANCE REQUIREMENTS

4.9.1.1 The more restrictive of the above two reactivity conditions shall be determined prior to:

a. Removing or unbolting the reactor vessel head, and

b. Withdrawal of any full-length control rod in excess of 3 feet from its fully inserted position within the reactor vessel.

4.9.1.2 The boron concentration of the Reactor Coolant System and the refueling canal shall be determined by chemical analysis at least once per 72 hours.

4.9.1.3 Primary water supply to the boric acid blender** shall be verified closed and secured in position by mechanical stops or by removal of air or electrical power at least once per 31 days.

4.9.1.4 The spent fuel pit boron concentration shall be determined at least once per 31 days.

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* The reactor shall be maintained in MODE 6 whenever fuel is in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

** Primary water supply to the boric acid blender may be opened under administrative controls for makeup.

3/4.9.2 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2 As a minimum, two Source Range Neutron Flux Monitors shall be OPERABLE, each with continuous visual indication in the control room and one with audible indication in the containment and control room.

APPLICABILITY: MODE 6.

ACTION:

a. With one of the above required monitors inoperable or not operating, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

b. With both of the above required monitors inoperable or not operating, determine the boron concentration of the Reactor Coolant System at least once per 12 hours.

SURVEILLANCE REQUIREMENTS

4.9.2 Each source Range Neutron Flux Monitor shall be demonstrated OPERABLE by performance of:

a. A CHANNEL CHECK at least once per 12 hours,

b. An ANALOG CHANNEL OPERATIONAL TEST within 8 hours prior to the initial start of CORE ALTERATIONS, and

c. An ANALOG CHANNEL OPERATIONAL TEST at least once per 7 days.

3/4.9.3 DECAY TIME

LIMITING CONDITION FOR OPERATION

3.9.3 The reactor shall be subcritical for at least 100 hours.

APPLICABILITY: During movement of irradiated fuel in the reactor vessel.

ACTION:

With the reactor subcritical for less than 100 hours, suspend all operations involving movement of irradiated fuel in the reactor vessel.

SURVEILLANCE REQUIREMENTS

4.9.3 The reactor shall be determined to have been subcritical for at least 100 hours by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor vessel.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

LIMITING CONDITION FOR OPERATION

3.9.4 The containment building penetrations shall be in the following status:

a. The equipment door closed and held in place by a minimum of four bolts,

b. A minimum of one door in each airlock is closed, and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:*

   1) Closed by an isolation valve or a manual valve, or a blind flange, or

   2) Be capable of being closed by an OPERABLE automatic containment ventilation isolation valve.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or movement of irradiated fuel in the containment building.

SURVEILLANCE REQUIREMENTS

4.9.4 Each of the above required containment building penetrations shall be determined to be either in its closed/isolated condition or capable of being closed by an OPERABLE automatic containment ventilation isolation valve, within 100 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS or movement of irradiated fuel in the containment building by:

a. Verifying the penetrations are in their closed/isolated condition, or

b. Testing the containment ventilation isolation valves per the applicable portion of Specification 4.6.4.2.

* Exception may be taken under Administrative Controls for opening of certain valves and airlocks necessary to perform surveillance or testing requirements.

3/4.9.5 COMMUNICATIONS

LIMITING CONDITION FOR OPERATION

3.9.5 Direct communications shall be maintained between the control room and personnel at the refueling station.

APPLICABILITY: During CORE ALTERATIONS.

ACTION:

When direct communications between the control room and personnel at the refueling station cannot be maintained, suspend all CORE ALTERATIONS.

SURVEILLANCE REQUIREMENTS

4.9.5 Direct communications between the control room and personnel at the refueling station shall be demonstrated within 1 hour prior to the start of and at least once per 12 hours during CORE ALTERATIONS.

3/4.9.6 MANIPULATOR CRANE

LIMITING CONDITION FOR OPERATION

3.9.6 The manipulator crane and auxiliary hoist shall be used for movement of drive rods or fuel assemblies and shall be OPERABLE with:

a. The manipulator crane used for movement of fuel assemblies having:

   1) A minimum capacity of 2750 pounds, and

   2) An overload cutoff limit less than or equal to 2700 pounds.

b. The auxiliary hoist used for latching and unlatching drive rods having:

   1) A minimum capacity of 610 pounds, and

   2) A load indicator which shall be used to indicate loads in excess of 600 pounds.

APPLICABILITY: During movement of drive rods or fuel assemblies within the reactor vessel.

ACTION:

With the requirements for crane and/or hoist OPERABILITY not satisfied, suspend use of any inoperable manipulator crane and/or auxiliary hoist from operations involving the movement of drive rods and fuel assemblies within the reactor vessel.

SURVEILLANCE REQUIREMENTS

4.9.6.1 Each manipulator crane used for movement of fuel assemblies within the reactor vessel shall be demonstrated OPERABLE within 100 hours prior to the start of such operations by performing a load test of at least 2750 pounds and demonstrating an automatic load cutoff when the crane load exceeds 2700 pounds. If operation is interrupted for 7 days the manipulator crane shall be retested.

4.9.6.2 Each auxiliary hoist and associated load indicator used for movement of drive rods within the reactor vessel shall be demonstrated OPERABLE within 100 hours prior to the start of such operations by performing a load test of at least 610 pounds. If operation is interrupted for 7 days the auxiliary hoist shall be retested.

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE AREAS

LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of 2000 pounds shall be prohibited from travel over fuel assemblies in the storage pool.*

APPLICABILITY: With fuel assemblies in the storage pool.

ACTION:

a. With the requirements of the above specifications not satisfied, place the crane load in a safe condition.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.7 Prior to crane operation over fuel assemblies in the spent fuel storage pool, verify that total loads are 2000 pounds or less.

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* The temporary construction crane to be used for the rerack operation may be carried over irradiated fuel to facilitate installation of the crane. Lift rigs which meet the design and operational requirements of NUREG 0612 "Control of Heavy Loads at Nuclear Power Plants" will be used while performing this installation.

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

HIGH WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.1 At least one residual heat removal (RHR) loop shall be OPERABLE and in operation.*

APPLICABILITY: MODE 6, when the water level above the top of the reactor vessel flange is greater than or equal to 23 feet.

ACTION:

With no RHR loop OPERABLE and in operation, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to OPERABLE and operating status as soon as possible. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.8.1.1 At least one RHR loop shall be verified in operation and circulating reactor coolant at a flow rate of greater than or equal to 3000 gpm at least once per 12 hours.

4.9.8.1.2 The RHR flow rate indicator shall be subjected to a CHANNEL CALIBRATION at least once per refueling.

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* The RHR loop may be removed from operation for up to 1 hour per 8-hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor vessel hot legs.

REFUELING OPERATIONS

LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.2 Two independent residual heat removal (RHR) loops shall be OPERABLE, and at least one RHR loop shall be in operation.*

APPLICABILITY: MODE 6, when the water level above the top of the reactor vessel flange is less than 23 feet.

ACTION:

a. With less than the required RHR loops OPERABLE, immediately initiate corrective action to return the required RHR loops to OPERABLE status, or establish greater than or equal to 23 feet of water above the reactor vessel flange, as soon as possible.

b. With no RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to operation. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.8.2 At least one RHR loop shall be verified in operation and circulating reactor coolant at a flow rate of greater than or equal to 3000 gpm at least once per 12 hours.

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* Prior to initial criticality, the RHR loop may be removed from operation for up to 1 hour per 8-hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor vessel hot legs.

3/4.9.9 CONTAINMENT VENTILATION ISOLATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.9 The Containment Ventilation Isolation System shall be OPERABLE.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

a. With the Containment Ventilation Isolation System inoperable, close each of the penetrations providing direct access from the containment atmosphere to the outside atmosphere.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.9 The Containment Ventilation Isolation System shall be demonstrated OPERABLE within 100 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS by verifying that containment ventilation isolation occurs on a High Radiation test signal from each of the containment process radiation monitoring instrumentation channels.

REFUELING OPERATIONS

3/4.9.10 WATER LEVEL - REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

3.9.10 At least 23 feet of water shall be maintained over the top of the reactor vessel flange.

APPLICABILITY: During movement of fuel assemblies or control rods within the containment when either the fuel assemblies being moved or the fuel assemblies seated within the reactor vessel are irradiated while in MODE 6.

ACTION:

With the requirements of the above specificiation not satisfied, suspend all operations involving movement of fuel assemblies or control rods within the reactor vessel.

SURVEILLANCE REQUIREMENTS

4.9.10 The water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours thereafter during movement of fuel assemblies or control rods.

3/4.9.11 WATER LEVEL - STORAGE POOL

LIMITING CONDITION FOR OPERATION

3.9.11 Water level at elevation 57' 0" (-2", + 12")* shall be maintained in the spent fuel storage pool.

APPLICABILITY: Whenever irradiated fuel assemblies are in the storage pool.

ACTION:

a. With the requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the fuel storage areas and restore the water level to within its limit within 4 hours.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.11 The water level in the storage pool shall be determined to be at least its minimum required depth at least once per 7 days when irradiated fuel assemblies are in the fuel storage pool.

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* During spent fuel rerack operation, the water level shall be maintained at least at 49' 0" elevation. There will be no movement of fuel assemblies with water level lower than 57'0" (-2", + 12") elevation during rerack operation.

3/4.9.12 HANDLING OF SPENT FUEL CASK

LIMITING CONDITION FOR OPERATION

3.9.12 The handling of spent fuel cask shall be limited to the following conditions:

1) The spent fuel cask shall not be moved into the spent fuel pit until all the spent fuel in the pit has decayed for a minimum of one thousand five hundred twenty-five (1,525) hours.*

2) Only a single element cask may be moved into the spent fuel pit.

3) A fuel assembly shall not be removed from the spent fuel pit in a shipping cask until it has decayed for a minimum of one hundred twenty (120) days.

APPLICABILITY: During movement of spent fuel cask in the spent fuel storage area.

ACTION:

With the requirement of the above specification not satisfied, suspend all movement of the spent fuel cask within the spent fuel storage area.

SURVEILLANCE REQUIREMENTS

4.9.12.1 The following required decay time of the spent fuel assemblies shall be determined prior to the movement of a spent fuel cask by verification of date and time the spent fuel assemblies were placed into the spent fuel pit:

a. 1525 hours of decay of all spent fuel assemblies in the spent fuel pit for movement of a spent fuel cask into the spent fuel pit.

b. 120 days of decay of the spent fuel assembly in the spent fuel cask prior to removal of the spent fuel cask from the spent fuel pit.

4.9.12.2 Prior to any operations involving spent fuel cask movement into the spent fuel pit, verify only a single element cask will be moved into the spent fuel pit.

4.9.12.3 The spent fuel cask crane interlock shall be demonstrated OPERABLE within 7 days of crane operation and at least once per 7 days (7 days is maximum time between tests; specification 4.0.2 does not apply here) when the crane is being used to maneuver the spent fuel cask.

* The spent fuel cask can be moved into the Unit 4 spent fuel pit after a minimum decay of 1000 hours until the new two-region high density spent fuel racks are installed.

3/4.9.13 RADIATION MONITORING

LIMITING CONDITION FOR OPERATION

3.9.13 The containment radiation monitors which intitate containment and control room ventilation isolation shall be OPERABLE.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

a) With one or both radiation monitors inoperable, operation may continue provided the containment ventilation isolation valves are maintained closed.

b) With one or both radiation monitors inoperable, within 1 hour isolate the Control Room Ventilation System and initiate operation of the Control Room Ventilation System in the recirculation mode.

SURVEILLANCE REQUIREMENTS

4.9.13 Each containment radiation monitor shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST at the frequencies shown in Table 4.3-3.

3/4.9.14 SPENT FUEL STORAGE

LIMITING CONDITION FOR OPERATION

3.9.14 The following conditions shall apply to spent fuel storage:

a. Fuel assemblies containing more than 4.1 weight percent of U-235 shall not be placed in the single region spent fuel storage racks. After installation of the two-region high density spent fuel racks, the maximum enrichment loading for fuel assemblies in the spent fuel racks shall be 4.5 weight percent of U-235.

b. The minimum boron concentration in the Spent Fuel Pit shall be 1950 ppm.

c.* Storage in Region II of the Spent fuel Pit shall be further restricted by burnup and enrichment limits specified in Table 3.9-1.

d.* During the re-racking operation only, fuel that does not meet the burnup requirement for normal storage in Region II may be stored in Region II in a checkerboard arrangment (i.e., no fuel stored in adjacent spaces).

APPLICABILITY: At all times when fuel is stored in the Spent Fuel Pit.

ACTION:

a. With any of conditions a, c or d not satisfied, suspend movement of additional fuel assemblies into the Spent Fuel Pit and restore the spent fuel storage configuration to within the specified conditions.

b. With boron concentration in the Spent Fuel Pit less than 1950 ppm, suspend movement of spent fuel in the Spent Fuel Pit and initiate action to restore boron concentration to 1950 ppm or greater.

SURVEILLANCE REQUIREMENTS

4.9.14 The boron concentration of the Spent Fuel Pit shall be verified to be 1950 ppm or greater at least once per month.

* These requirements are applicable only after installation of the new two-region high density spent fuel racks.

TABLE 3.9-1

SPENT FUEL BURNUP REQUIREMENTS FOR STORAGE IN REGION II OF THE SPENT FUEL PIT

Initial w/o	Discharge Burnup GWD/MT
1.5	0
1.75	5.0
2.0	9.0
2.2	12.0
2.4	14.8
2.6	17.6
2.8	20.1
3.0	22.6
3.2	25.0
3.4	27.4
3.6	29.6
3.8	31.8
4.0	34.0
4.2	36.1
4.5	39.0

Linear interpolation between two consecutive points will yield conservative results.

3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of control rod worth and SHUTDOWN MARGIN provided shutdown reactivity equivalent to at least the highest worth control rod is available for trip insertion from OPERABLE control rod(s).

APPLICABILITY: MODE 2.

ACTION:

a. With any full-length control rod not fully inserted and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at greater than or equal to 10 gpm of a solution containing greater than or equal to 20,000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full-length control rods fully inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at greater than or equal to 10 gpm of a solution containing greater than or equal to 20,000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full-length control rod either partially or fully withdrawn shall be determined at least once per 2 hours.

4.10.1.2 Each full-length control rod not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 24 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

SPECIAL TEST EXCEPTIONS

3/4.10.2 GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The group height, insertion, and power distribution limits of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1 and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is maintained less than or equal to 85% of RATED THERMAL POWER, and

b. The limits of Specifications 3.2.2 and 3.2.3 are maintained and determined at the frequencies specified in Specification 4.10.2.2 below.

APPLICABILITY: MODE 1

ACTION:

With any of the limits of Specification 3.2.2 or 3.2.3 being exceeded while the requirements of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1 and 3.2.4 are suspended, either:

a. Reduce THERMAL POWER sufficient to satisfy the ACTION requirements of Specifications 3.2.2 and 3.2.3, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined to be less than or equal to 85% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.2.2 The requirements of the below listed specifications shall be performed at least once per 12 hours during PHYSICS TESTS:

a. Specifications 4.2.2.1 or 4.2.2.2

b. Specification 4.2.3.2.

SPECIAL TEST EXCEPTIONS

3/4.10.3 PHYSICS TESTS

LIMITING CONDITION FOR OPERATION

3.10.3 The limitations of Specifications 3.1.1.3, 3.1.1.4, 3.1.3.1, 3.1.3.5, and 3.1.3.6 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER,

b. The Reactor Trip Setpoints on the OPERABLE Intermediate and Power Range channels are set at less than or equal to 25% of RATED THERMAL POWER, and

c. The Reactor Coolant System lowest operating loop temperature (T_{avg}) is greater than or equal to 531 F.

APPLICABILITY: MODE 2.

ACTION:

a. With the THERMAL POWER greater than 5% of RATED THERMAL POWER, immediately open the Reactor trip breakers.

b. With a Reactor Coolant System operating loop temperature (T_{avg}) less than 531 F, restore T_{avg} to within its limit within 15 minutes or be in at least HOT STANDBY within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.10.3.1 The THERMAL POWER shall be determined to be less than or equal to 5% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.3.2 Each Intermediate and Power Range channel shall be subjected to an ANALOG CHANNEL OPERATIONAL TEST within 12 hours prior to initiating PHYSICS TESTS.

4.10.3.3 The Reactor Coolant System temperature (T_{avg}) shall be determined to be greater than or equal to 531 F at least once per 30 minutes during PHYSICS TESTS.

SPECIAL TEST EXCEPTIONS

3/4.10.4 REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

3.10.4 The limitations of Specification 3.4.1.1 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER, and

b. The Reactor Trip Setpoints on the OPERABLE Intermediate and Power Range channels are set less than or equal to 25% of RATED THERMAL POWER.

APPLICABILITY: MODE 2

ACTION:

With the THERMAL POWER greater than 5% of RATED THERMAL POWER, immediately open the Reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The THERMAL POWER shall be determined to be less than 5% RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.4.2 Each Intermediate and Power Range channel shall be subjected to an ANALOG CHANNEL OPERATIONAL TEST within 12 hours prior to initiating PHYSICS TESTS.

SPECIAL TEST EXCEPTIONS

3/4.10.5 POSITION INDICATION SYSTEM - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.10.5 The limitations of Specification 3.1.3.3 may be suspended during the performance of individual full-length shutdown and control rod drop time measurements provided:

a. Only one shutdown or control bank is withdrawn from the fully inserted position at a time, and

b. The rod position indicator is OPERABLE during the withdrawal of the rods.

APPLICABILITY: MODES 3, 4 and 5 during performance of rod drop time measurements.

ACTION:

With the Position Indication Systems inoperable or with more than one bank of rods withdrawn, immediately open the Reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.5 The above required Position Indication Systems shall be determined to be OPERABLE within 24 hours prior to the start of and at least once per 24 hours thereafter during rod drop time measurements by verifying the Demand Position Indication System and the Individual Rod Position Indication System agree:

a. Within 12 steps when the rods are stationary, and

b. Within 24 steps during rod motion.

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.11.1.1 The concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS (see Figure 5.1-1) shall be limited to the concentrations specified in 10 CFR Part 20, Appendix B, Table II, Column 2 for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall be limited to 2×10^{-4} microCurie/ml total activity.

APPLICABILITY: At all times.

ACTION:

With the concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS exceeding the above limits, immediately restore the concentration to within the above limits.

SURVEILLANCE REQUIREMENTS

4.11.1.1.1 Radioactive liquid wastes shall be sampled and analyzed according to the sampling and analysis program of Table 4.11-1.

4.11.1.1.2 The results of the radioactivity analyses shall be used in accordance with the methodology and parameters in the ODCM to assure that the concentrations at the point of release are maintained within the limits of Specification 3.11.1.1.

TABLE 4.11-1

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

LIQUID RELEASE TYPE	SAMPLING FREQUENCY	MINIMUM ANALYSIS FREQUENCY	TYPE OF ACTIVITY ANALYSIS	LOWER LIMIT OF DETECTION (LLD)(1) (μCi/ml)
1. Batch Waste Release Tanks (2)	P Each Batch	P Each Batch	Principal Gamma Emitters (3)	5×10^{-7}
			I-131	1×10^{-6}
	P One Batch/M	M	Dissolved and Entrained Gases (Gamma Emitters)	1×10^{-5}
	P Each Batch	M Composite (4)	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	P Each Batch	Q Composite (4)	Sr-89,Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}
2. Continuous Releases (5)				
a. Steam Generator Blowdown (7)	4/M	4/M	Principal Gamma Emitters (3)	5×10^{-7}
			I-131	1×10^{-6}
	M(8)	M(8)	Dissolved and Entrained Gases (Gamma Emitters)	1×10^{-5}
	4/M (8)	M(8) Composite (6)	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	4/M (8)	Q(8) Composite (6)	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}
b. Storm Drain	M	M	Principal Gamma Emitters (3)	5×10^{-7}
			I-131	1×10^{-6}

TABLE 4.11-1 (Continued)

TABLE NOTATIONS

(1) The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

$$LLD = \frac{4.66\ s_b}{(E)(V)(2.22 \times 10^6)(Y)[\exp(-\lambda \Delta t)]}$$

Where:

LLD = the "a priori" lower limit of detection (μCurie per unit) mass or volume,

s_b = the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (counts per minute),

E = the counting efficiency (counts per disintegration),

V = the sample size (units of mass or volume),

$2.22 \times 10^{+6}$ = the number of disintegrations per minute per μCurie,

Y = the fractional radiochemical yield, when applicable,

λ = the radioactive decay constant for the particular radionuclide and

Δt = the elapsed time between the midpoint of sample collection and the time of counting.

Typical values of E, V, Y and Δt should be used in the calculation.

It should be recognized that the LLD is defined as an <u>a priori</u> (before the fact) limit representing the capability of a measurement system and not as an <u>a posteriori</u> (after the fact) limit for a particular measurement.

(2) A batch release is the discharge of liquid wastes of a discrete volume. Prior to sampling for analyses, each batch shall be isolated, and then thoroughly mixed to assure representative sampling.

TABLE 4.11-1 (Continued)

TABLE NOTATIONS

(3) The principal gamma emitters for which the LLD specification exclusively applies are the following radionuclides: Mn-54, FE-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141, and Ce-144. This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.4.

(4) A composite sample is one in which the quantity of liquid sampled is proportional to the quantity of liquid waste discharged and in which the method of sampling employed results in a specimen that is representative of the liquids released.

(5) A continuous release is the discharge of liquid wastes of a nondiscrete volume, e.g., from a volume of a system that has an input flow during the continuous release.

(6) Prior to analyses, all samples taken for the composite shall be thoroughly mixed in order for the composite sample to be representative of the effluent release.

(7) Sampling and analysis of steam generator blowdown is not required during Mode 5 or 6.

(8) Sampling and analysis of steam generator blowdown on the applicable Unit for these species, is only necessary when primary to secondary leakage is occuring as indicated by the condenser air ejector monitor.

RADIOACTIVE EFFLUENTS

DOSE

LIMITING CONDITION FOR OPERATION

3.11.1.2 The dose or dose commitment to a MEMBER OF THE PUBLIC from raioactive materials in liquid effluents released, from each unit, to UNRESTRICTED AREAS (see Figure 5.1-1) shall be limited:

a. During any calender quarter to less than or equal to 1.5 mrems to the whole body and to less than or equal to 5 mrems to any organ, and

b. During any calendar year to less than or equal to 3 mrems to the whole body and to less than or equal to 10 mrems to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.2 Cumulative dose contributions from liquid effluents for the current calendar quarter and the current calendar year shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

RADIOACTIVE EFFLUENTS

LIQUID RADWASTE TREATMENT SYSTEM

LIMITING CONDITION FOR OPERATION

3.11.1.3 The Liquid Radwaste Treatment System shall be OPERABLE and appropriate portions of the system shall be used to reduce releases of radioactivity when the projected doses due to the liquid effluent, from each unit, to UNRESTRICTED AREAS (see Figure 5.1-1) would exceed 0.06 mrem to the whole body or 0.2 mrem to any organ when averaged over a 31-day period.

APPLICABILITY: At all times.

ACTION:

a. With radioactive liquid waste being discharged without treatment and in excess of the above limits and any portion of the Liquid Radwaste Treatment System not in operation, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that includes the following information:

    1. Explanation of why liquid radwaste was being discharged without treatment, identification of any inoperable equipment or subsystems, and the reason for the inoperability,

    2. Action(s) taken to restore the inoperable equipment to OPERABLE status, and

    3. Summary description of action(s) taken to prevent a recurrence.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.3.1 Doses due to liquid releases from each unit to UNRESTRICTED AREAS shall be projected at least once per 31 days in accordance with the methodology and parameters in the ODCM when Liquid Radwaste Treatment Systems are not being fully utilized.

4.11.1.3.2 The installed Liquid Radwaste Treatment System shall be considered OPERABLE by meeting Specifications 3.11.1.1 and 3.11.1.2.

RADIOACTIVE EFFLUENTS

3/4.11.2 GASEOUS EFFLUENTS

DOSE RATE

LIMITING CONDITION FOR OPERATION

3.11.2.1 The dose rate due to radioactive materials released in gaseous effluents from the site to areas at and beyond the SITE BOUNDARY (see Figure 5.1-1) shall be limited to the following:

a. For noble gases: Less than or equal to 500 mrems/yr to the whole body and less than or equal to 3000 mrems/yr to the skin, and

b. For Iodine-131, for Iodine-133, for tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: Less than or equal to 1500 mrems/yr to any organ.

APPLICABILITY: At all times.

ACTION:

With the dose rate(s) exceeding the above limits, immediately restore the release rate to within the above limit(s).

SURVEILLANCE REQUIREMENTS

4.11.2.1.1 The dose rate due to noble gases in gaseous effluents shall be determined to be within the above limits in accordance with the methodology and parameters in the ODCM.

4.11.2.1.2 The dose rate due to Iodine-131, Iodine-133, tritium and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents shall be determined to be within the above limits in accordance with the methodology and parameters in the ODCM by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified in Table 4.11-2.

TABLE 4.11-2

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

GASEOUS RELEASE TYPE	SAMPLING FREQUENCY	MINIMUM ANALYSIS FREQUENCY	TYPE OF ACTIVITY ANALYSIS	LOWER LIMIT OF DETECTION (LLD) (1) (μCi/CC)
1. Gas Decay Tank (Batch)	P Each Tank Grab Sample	P Each Tank	Principal Gamma Emitters(2)	$1x10^{-4}$
2. Containment PURGE or VENTING (Batch)	P(6) Grab Sample	P(6)	Principal Gamma Emitters(2)	$1x10^{-4}$
			H-3	$1x10^{-6}$
3. Condenser Air Ejectors	M(6) Grab Sample	M(6) Gas Sample	Principal Gamma Emitters(2)	$1x10^{-4}$
			H-3	$1x10^{-6}$
	Continuous(3)	4/M Charcoal Sample	1-131	$1x10^{-12}$
	Continuous(3)	4/M Particulate Sample	Principal Gamma Emitters(2)	$1x10^{-11}$
4. Plant Vent (The Plant Vent incorporates Unit 4 Spent Fuel Pit Building Vent)	M(6) Grab Sample	M(6) Gas Sample	Principal Gamma Emitters(2)	$1x10^{-4}$
	M(4)(5) Grab Sample	M	H-3	$1x10^{-6}$
	Continuous(3)	4/M (7) Charcoal Sample	I-131	$1x10^{-12}$

TABLE 4.11-2
(Continued)

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

4. (cont.)	Continuous(3)	$4/M^{(7)}$ Particulate Sample	Principal Gamma Emitters(2)	$1x10^{-11}$
	Continuous(3)	M Composite Particulate Sample	Gross Alpha	$1x10^{-11}$
	Continuous(3)	Q Composite Particulate Sample	Sr-89,Sr-90	$1x10^{-11}$
5. Unit 3 Spent Fuel Pit Building Vent	M Grab Sample	M Gas Sample	Principal Gamma Emitters(2)	$1x10^{-4}$
	$M^{(4)(5)}$ Grab Sample	M	H-3	$1x10^{-6}$
	Continuous(3)	4/M Charcoal Sample	I-131	$1x10^{-12}$
	Continuous(3)	4/M Particulate Sample	Principal Gamma Emitters(2)	$1x10^{-11}$
	Continuous(3)	M Composite Particulate Sample	Gross Alpha	$1x10^{-11}$
	Continuous(3)	Q Composite Particulate Sample	Sr-89,Sr-90	$1x10^{-11}$
All release Types for 3, 4 and 5	Continuous(3)	Noble Gas Monitor	Noble Gases Gross Beta or Gamma	$1x10^{-6}$

TABLE 4.11-2
(Continued)

TABLE NOTATION

1. The LLD is the smallest concentration of radioactive material in a sample that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system (which may include radiochemical separation):

$$LLD = \frac{4.66\, s_b}{(E)(V)(2.22 \times 10^6)(Y)[\exp(-\lambda\Delta t)]}$$

Where:

LLD = the "a priori" lower limit of detection as defined above for a blank sample (as μCurie per unit mass or volume),

s_b = the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (counts per minute),

E = the counting efficiency (counts per transformation),

V = the sample size (units of mass or volume),

$2.22 \times 10^{+6}$ = the number of transformations per minute per μCurie,

Y = the fractional radiochemical yield, when applicable,

λ = the radioactive decay constant for the particular radionuclide and

Δt = the elapsed time between the midpoint of sample collection and the time of counting (for plant effluents, not environmental samples).

The value of s_b used in the calculation of the LLD for a detection system shall be based on the actual observed variance of the background counting rate or of the counting rate of the blank samples (as appropriate) rather than on an unverified theoretically predicted variance. Typical values of E, V, Y and Δt shall be used in the calculation.

TABLE 4.11-2
(Continued)

TABLE NOTATIONS

2. The principal gamma emitters for which the LLD limit specification will apply are exclusively the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 for noble gas emissions and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141, Ce-144 and I-131 for particulate emissions. This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measurable and identifiable, together with the above nuclides shall also be identified and reported pursuant to Specification 6.9.1.4. Nuclides which are below the LLD for the analyses should not be reported as being present at the LLD level for that nuclide. When a radionuclide's calculated LLD is greater than its listed LLD limit, the calculated LLD should be assigned as the activity of the radionuclide; or, the activity of the radionuclide should be calculated using measured ratios with those radionuclides which are routinely identified and measured.

3. The ratio of the sample flow rate to the sampled stream flow rate shall be known for the time period covered by each dose or dose rate calculation made in accordance with Specifications 3.11.2.1, 3.11.2.2 and 3.11.2.3.

4. Tritium grab samples shall be taken on the applicable Unit at least once per 24 hours when the refueling canal is flooded.

5. Tritium grab samples shall be taken on the applicable Unit at least 4 times per month from the ventilation exhaust from the spent fuel pool area, whenever spent fuel is in the spent fuel pool.

6. Sampling and analysis on the applicable Unit shall also be performed following shutdown, startup or a THERMAL POWER change exceeding 15% of RATED THERMAL POWER within 1 hour if (1) analysis shows that the DOSE EQUIVALENT I-131 concentration in the primary coolant has increased by more than a factor of 3; or (2) the noble gas activity monitor shows that effluent activity has increased by more than a factor of 3.

7. Sample collection media on the applicable Unit shall be changed at least 4 times per month and analyses shall be completed within 48 hours after changing (or after removal from sampler). Sampling shall also be performed at least once per 24 hours for at least 7 days following each shutdown, startup or THERMAL POWER change exceeding 15% of RATED THERMAL POWER in 1 hour and analyses shall be completed within 48 hours of changing if (1) analysis shows that the DOSE EQUIVALENT I-131 concentration in the primary coolant has increased by more than a factor of 3; and (2) the noble gas activity monitor shows that effluent activity has increased by more than a factor of 3. When samples collected for 24 hours are analyzed, the corresponding LLDs may be increased by a factor of 10.

RADIOACTIVE EFFLUENTS

DOSE-NOBLE GASES

LIMITING CONDITION FOR OPERATION

3.11.2.2 The air dose due to noble gases released in gaseous effluents, from each unit, to areas at and beyond the SITE BOUNDARY (see Figure 5.1-1) shall be limited to the following:

a. During any calendar quarter: Less than or equal to 5 mrads for gamma radiation and less than or equal to 10 mrads for beta radiation, and

b. During any calendar year: Less than or equal to 10 mrads for gamma radiation and less than or equal to 20 mrads for beta radiation.

APPLICABILITY: At all times.

ACTION:

a. With the calculated air dose from radioactive noble gases in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.2 Cumulative dose contributions for the current calendar quarter and current calendar year for noble gases shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

RADIOACTIVE EFFLUENTS

DOSE - IODINE-131, IODINE-133, TRITIUM, AND RADIOACTIVE MATERIAL IN PARTICULATE FORM

LIMITING CONDITION FOR OPERATION

3.11.2.3 The dose to a MEMBER OF THE PUBLIC from Iodine-131, Iodine-133, tritium and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released, from each unit, to areas at and beyond the SITE BOUNDARY (see Figure 5.1-1) shall be limited to the following:

a. During any calendar quarter: Less than or equal to 7.5 mrems to any organ and,

b. During any calendar year: Less than or equal to 15 mrems to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of Iodine-131, Iodine-133, tritium, and radionuclides in particulate form with half-lives greater than 8 days, in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.3 Cumulative dose contributions for the current calendar quarter and current calendar year for Iodine-131, Iodine-133, tritium and radionuclides in particulate form with half-lives greater than 8 days shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

GASEOUS RADWASTE TREATMENT SYSTEM

LIMITING CONDITION FOR OPERATION

3.11.2.4 The VENTILATION EXHAUST TREATMENT SYSTEM and the GAS DECAY TANK SYSTEM shall be OPERABLE and appropriate portions of these systems shall be used to reduce releases of radioactivity when the projected doses in 31 days due to gaseous effluent releases, from each unit, to areas at and beyond the SITE BOUNDARY (see Figure 5.1-1) would exceed:

a. 0.2 mrad to air from gamma radiation, or

b. 0.4 mrad to air from beta radiation, or

c. 0.3 mrem to any organ of a MEMBER OF THE PUBLIC.

APPLICABILITY: At all times.

ACTION:

a. With radioactive gaseous waste being discharged without treatment and in excess of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that includes the following information:

   1. Identification of any inoperable equipment or subsystems, and the reason for the inoperability,

   2. Action(s) taken to restore the inoperable equipment to OPERABLE status, and

   3. Summary description of action(s) taken to prevent a recurrence.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.4.1 Doses due to gaseous releases from each unit to areas at and beyond the SITE BOUNDARY shall be projected at least once per 31 days in accordance with the methodology and parameters in the ODCM when Gaseous Radwaste Treatment Systems are not being fully utilized.

4.11.2.4.2 The installed VENTILATION EXHAUST TREATMENT SYSTEM and GAS DECAY TANK SYSTEM shall be considered OPERABLE by meeting Specifications 3.11.2.1 and 3.11.2.2 or 3.11.2.3.

RADIOACTIVE EFFLUENTS

EXPLOSIVE GAS MIXTURE

LIMITING CONDITION FOR OPERATION

3.11.2.5 The concentration of oxygen in the GAS DECAY TANK SYSTEM shall be limited to less than or equal to 2% by volume whenever the hydrogen concentration exceeds 4% by volume.

APPLICABILITY: At all times.

ACTION:

a. With the concentration of oxygen in the in-service GAS DECAY TANK greater than 2% by volume but less than or equal to 4% by volume, reduce the oxygen concentration to the above limits within 48 hours.

b. With the concentration of oxygen in the in-service GAS DECAY TANK greater than 4% by volume and the hydrogen concentration greater than 2% by volume, immediately suspend all additions of waste gases to the affected GAS DECAY TANKS and reduce the concentration of oxygen to less than or equal to 4% by volume, then take ACTION a., above.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.5 The concentrations of hydrogen and oxygen in the in-service GAS DECAY TANK shall be determined to be within the above limits by continously* monitoring the waste gases in the in-service GAS DECAY TANK with the hydrogen and oxygen monitors required OPERABLE by Table 3.3-8 of Specification 3.3.3.7.

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* Until modifications to the gas analyzing equipment are complete, monitoring will consist of grab sampling and analysis at the following frequencies:

a) Once per 8 hours during degassing operations.

b) Once per day during other operations.

RADIOACTIVE EFFLUENTS

GAS DECAY TANKS

LIMITING CONDITION FOR OPERATION

3.11.2.6 The quantity of radioactivity contained in each gas decay tank shall be limited to less than or equal to 70,000 Curies of noble gases (considered as Xe-133 equivalent).

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any gas decay tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank, within 48 hours reduce the tank contents to within the limit, and describe the events leading to this condition in the next Semiannual Radioactive Effluent Release Report, pursuant to Specification 6.9.1.4.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE RQUIREMENTS

4.11.2.6 The quantity of radioactive material contained in each gas decay tank shall be determined to be within the above limit at least once per 24 hours when radioactive materials are being added to the tank and the Reactor Coolant System total activity exceeds the limit of Specification 3.4.8.

3/4.11.3 SOLID RADIOACTIVE WASTES

LIMITING CONDITION FOR OPERATION

3.11.3 Radioactive wastes shall be solidified or dewatered in accordance with the PROCESS CONTROL PROGRAM to meet shipping and transportation requirements during transit, and disposal site requirements when received at the disposal site.

APPLICABILITY: At all times.

ACTION:

a. With SOLIDIFICATION or dewatering not meeting disposal site and shipping and transportation requirements, suspend shipment of the inadequately processed wastes and correct the PROCESS CONTROL PROGRAM, the procedures, and/or the Solid Waste System as necessary to prevent recurrence.

b. With SOLIDIFICATION or dewatering not performed in accordance with the PROCESS CONTROL PROGRAM, test the improperly processed waste in each container to ensure that it meets burial ground and shipping requirements and take appropriate administrative action to prevent recurrence.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.3 SOLIDIFICATION* of at least one representative test specimen from at least every tenth batch of each type of wet radioactive wastes (e.g., filter sludges, spent resins, evaporator bottoms, boric acid solutions, and sodium sulfate solutions) shall be verified in accordance with the PROCESS CONTROL PROGRAM:

a. If any test specimen fails to verify SOLIDIFICATION, the SOLIDIFICATION of the batch under test shall be suspended until such time as additional test specimens can be obtained, alternative SOLIDIFICATION parameters can be determined in accordance with the PROCESS CONTROL PROGRAM, and a subsequent test verifies SOLIDIFICATION. SOLIDIFICATION of the batch may then be resumed using the alternative SOLIDIFICATION parameters determined by the PROCESS CONTROL PROGRAM;

* Does not include the process of dewatering.

SURVEILLANCE REQUIREMENTS (Continued)

b. If the initial test specimen from a batch of waste fails to verify SOLIDIFICATION, the PROCESS CONTROL PROGRAM shall provide for the collection and testing of representative test specimens from each consecutive batch of the same type of wet waste until at least three consecutive initial test specimens demonstrate SOLIDIFICAION. The PROCESS CONTROL PROGRAM shall be modified as required, as provided in Specification 6.13, to assure SOLIDIFICATION of subsequent batches of waste; and

c. With the installed equipment incapable of meeting Specification 3.11.3 or declared inoperable, restore the equipment to OPERABLE status or provide for contract capability to process waste as necessary to satisfy all applicable transportation and disposal requirements.

RADIOACTIVE EFFLUENTS

3/4.11.4 TOTAL DOSE

LIMITING CONDITION FOR OPERATION

3.11.4 The annual (calendar year) dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources shall be limited to less than or equal to 25 mrems to the whole body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrems.

APPLICABILITY: at all times.

ACTION:

a. With the calculated doses from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limit of specification 3.11.1.2a, 3.11.1.2b, 3.11.2.2a, 3.11.2.2b, 3.11.2.3a or 3.11.2.3b, calculations shall be made including direct radiation contributions from the units to determine whether the above limits of Specifications 3.11.4 have been exceeded. If such is the case, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that defines the corrective action to be taken to reduce subsequent releases to prevent recurrence of exceeding the above limits and includes the schedule for achieving conformance with the above limits. This Special Report, as defined in 10 CFR 20.405(c), shall include an analysis that estimates the radiation exposure (dose) to a MEMBER OF THE PUBLIC from uranium fuel cycle sources, including all effluent pathways and direct radiation, for the calendar year that includes the release(s) covered by this report. It shall also describe levels of radiation and concentrations of radioactive material involved, and the cause of the exposure levels or concentrations. If the estimated dose(s) exceeds the above limits, and if the release condition resulting in violation of 40 CFR Part 190 has not already been corrected, the Special Report shall include a request for a variance in accordance with the provisions of 40 CFR Part 190. Submittal of the report is considered a timely request, and a variance is granted until staff action on the request is complete.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.4.1 Cumulative dose contributions from liquid and gaseous effluents shall be determined in accordance with Specifications 4.11.1.2, 4.11.2.2, and 4.11.2.3, and in accordance with the methodology and parameters in the ODCM.

4.11.4.2 Cumulative dose contributions from direct radiation from the units and the methodology used shall be indicated in the Semiannual Radioactive Effluent Release Report. This requirement is applicable only under conditions set forth in ACTION a. of Specification 3.11.4.

3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.1 MONITORING PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.1 The Radiological Environmental Monitoring program shall be conducted as specified in Table 3.12-1.

APPLICABILITY: At all times.

ACTION:

a. With the Radiological Environmental Monitoring Program not being conducted as specified in Table 3.12-1, prepare and submit to the Commission, in the Annual Radiological Environmental Operating Report required by Specification 6.9.1.3, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence.

b. With the level of confirmed* radioactivity as the result of plant effluents in an environmental sampling medium at a specified location exceeding the reporting levels of Table 3.12-2 when averaged over any calendar quarter, prepare and submit to the Commission within 30 days, pursuant to specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to reduce radioactive effluents so that the potential annual dose** to a MEMBER OF THE PUBLIC is less than the calendar year limits of Specifications 3.11.1.2, 3.11.2.2, or 3.11.2.3. When more than one of the radionuclides in Table 3.12-2 are detected in the sampling medium, this report shall be submitted if:

$$\frac{\text{concentration (1)}}{\text{reporting level (1)}} + \frac{\text{concentration (2)}}{\text{reporting level (2)}} + \ldots \geq 1.0$$

When radionuclides other than those in Table 3.12-2 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose** to a MEMBER OF THE PUBLIC from all radonuclides is equal to or greater than the calendar year limits of Specifications 3.11.1.2, 3.11.2.2, or 3.11.2.3. This report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, the condition shall be reported and described in the Annual Radiological Environmental Operating Report required by Specification 6.9.1.3.

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* A confirmatory reanalysis of the original, a duplicate, or a new sample may be desirable, as appropriate. The results of the confirmatory analysis shall be completed at the earliest time consistent with the analysis, but in any case within 30 days.

** The methodology and parameters used to estimate the potential annual dose to a MEMBER OF THE PUBLIC shall be indicated in this report.

RADIOLOGICAL ENVIRONMENTAL MONITORING

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

c. With milk or broad leaf vegetation samples unavailable from one or more of the sample locations required by Table 3.12-1, identify specific locations for obtaining replacement samples and add them within 30 days to the Radiological Environmental Monitoring Program given in the ODCM. The specific locations from which samples were unavailable may then be deleted from the monitoring program. Pursuant to Specification 6.14, submit in the next Semiannual Radioactive Effluent Release Report documentation for a change in the ODCM including a revised figure(s) and table for the ODCM reflecting the new location(s) with supporting information identifying the cause of the unavailability of samples and justifying the selection of the new location(s) for obtaining samples.

d. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.1 The radiological environmental monitoring samples shall be collected pursuant to Table 3.12-1 from the specific locations given in the table and figure(s) in the ODCM, and shall be analyzed pursuant to the requirements of Table 3.12-1 and the detection capabilities required by Table 4.12-1.

TABLE 3.12-1

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM [1]

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS[2][3]	SAMPLING AND COLLECTION FREQUENCY [4]	TYPE AND FREQUENCY OF ANALYSIS [4]
1. DIRECT RADIATION[5]	21 Locations	Continuous monitoring with sample collection quarterly[6]	Gamma exposure rate - quarterly
2. AIRBORNE			
Radioiodine and Particulates	5 Locations	Continuous sampler operation with sample collection <u>weekly</u>, or more frequently if required by dust loading	<u>Radioiodine Filter</u>: 1-131 analysis weekly <u>Particulate Filter</u>: Gross beta radioactivity analysis $\geq$ 24 hours following a filter change [7] Gamma isotopic [8] analysis of composite [7] (by location) quarterly

TABLE 3.12-1
(Continued)
RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM (1)

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS (2)(3)	SAMPLING AND COLLECTION FREQUENCY (4)	TYPE AND FREQUENCY OF ANALYSIS (4)
3. WATERBONE (10)			
a. Surface (8)	3 Locations (9)	Monthly	Gamma isotopic (8) and tritium analysis monthly
b. Sediment from shoreline	3 Locations	Semiannually	Gamma isotopic (8) analysis semiannually
4. INGESTION			
a. Fish and Invertebrates			
1. Crustacea	2 Locations	Semiannually	Gamma isotopic (8) analysis semiannualy

TABLE 3.12-1
(Continued)
RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM [1]

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS[2][3]	SAMPLING AND COLLECTION FREQUENCY [4]	TYPE AND FREQUENCY OF ANALYSIS [4]
2. Fish	2 Locations	Semiannually	Gamma isotopic[8] analysis semiannually
b. Food Products			
1. Broad leaf vegetation	3 Locations[11]	Monthly when available	Gamma isotopic[8] and I-131 analysis monthly

TABLE 3.12-1
(Continued)

TABLE NOTATION

(1) Deviations are permitted from the required sampling schedule if specimens are unobtainable due to hazardous conditions, seasonal unavailability, malfunction of automatic sampling equipment or other legitimate reasons. If specimens are unobtainable due to sampling equipment malfunction, corrective action shall be taken prior to the end of the next sampling period. All deviations from the sampling schedule shall be documented in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.3.

(2) Specific parameters of distance and direction sector from the centerline of the plant vent stack, and additional description where pertinent, shall be provided for each sample location in Table 3.12-1 in a Table and figure(s) in the ODCM.

(3) At times, it may not be possible or practicable to continue to obtain samples of the media of choice at the most desired location or time. In these instances suitable alternative media and locations may be chosen for the particular pathway in question and appropriate substitutions made within 30 days in the radiological environmental monitoring program.

(4) The following definition of frequencies shall apply to Table 3.12-1 only:

Weekly - Not less than once per calendar week. A maximum interval of 11 days is allowed between the collection of any two consecutive samples.

Semi-Monthly - Not less than 2 times per calendar month with an interval of not less than 7 days between sample collections. A maximum interval of 24 days is allowed between collection of any two consecutive samples.

Monthly - Not less than once per calendar month with an interval of not less than 10 days between collection of any two consecutive samples.

Quarterly - Not less than once per calendar quarter.

Semiannualy - One sample each between calendar dates (January 1 - June 30) and (July 1 - December 31). An interval of not less than 30 days will be provided between sample collections.

The frequency of analyses is to be consistent with the sample collection frequency.

TABLE 3.12-1
(Continued)

TABLE NOTATION

(5) One or more instruments, such as a pressurized ion chamber, for measuring and recording dose rate continuously may be used in place of, or in addition to, integrating dosimeters. For purposes of this table, a thermoluminescent dosimeter (TLD) is considered to be one phosphor; two or more phosphors in a packet are considered as two or more dosimeters.

(6) Refers to normal collection frequency. More frequent sample collection is permitted when conditions warrant it.

(7) Airborne particulate sample filters are analyzed for gross beta radioactivity 24 hours or more after sampling to allow for radon and thoron daughter decay. In addition to the requirement for a gamma isotopic on a composite sample, a gamma isotopic is also required for each sample having a gross beta radioactivity which is > 1.0 pCi/m^3 and which is also > 10 times that of the most recent control sample.

(8) Gamma isotopic analysis means the identification and quantification of gamma-emitting radionuclides that may be attributable to the effluents from the facility.

(9) Off-shore grab samples.

(10) Discharges from the Turkey Point Plant do not influence drinking water or ground water pathways.

(11) Samples of broad leaf vegetation grown nearest each of two different offsite locations of highest predicted annual average ground level D/Q, and one sample of similar broad leaf vegetation at an available location 15-30 km distant in the least prevalent wind direction based upon historical data in the ODCM.

TABLE 3.12-2

REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

REPORTING LEVELS

ANALYSIS	WATER (pCi/ℓ)	AIRBORNE PARTICULATE OR GAS (pCi/m3)	FISH (pCi/kg, wet)	MILK (pCi/ℓ)	FOOD PRODUCTS (pCi/kg, wet)
H-3	30,000*				
Mn-54	1,000		30,000		
Fe-59	400		10,000		
Co-58	1,000		30,000		
Co-60	300		10,000		
Zn-65	300		20,000		
Zr-Nb-95***	400				
I-131	2**	0.9		3	100
Cs-134	30	10	1,000	60	1,000
Cs-137	50	20	2,000	70	2,000
Ba-La-140***	200			300	

* Since no drinking water pathway exists, a value of 30,000 pCi/ℓ is used. For drinking water sample, a value of 20,000 pCi/ℓ is used. This is 40 CFR Part 141 value.

** Applies to drinking water.

*** An equilibrium mixture of the parent and daughter isotopes which corresponds to the reporting value of the parent isotope.

TABLE 4.12-1

DETECTION CAPABILITIES FOR ENVIRONMENTAL SAMPLE ANALYSIS[1]

LOWER LIMIT OF DETECTION (LLD)[2][3]

ANALYSIS	WATER (pCi/ℓ)	AIRBORNE PARTICULATE OR GAS (pCi/m3)	FISH (pCi/kg, wet)	MILK (pCi/ℓ)	FOOD PRODUCTS (pCi/kg, wet)	SEDIMENT (pCi/kg, dry)
Gross beta	4	0.01				
H-3	3,000*					
Mn-54	15		130			
Fe-59	30		260			
Co-58,60	15		130			
Zn-65	30		260			
Zr-Nb-95	15 (5)					
I-131	1 (4)	0.07		1	60	
Cs-134	15	0.05	130	15	60	150
Cs-137	18	0.06	150	18	80	180
Ba-La-140	15 (5)			15 (5)		

* Since no drinking water pathway exists, a value of 3,000 pCi/ℓ is used.
For drinking water sample, a value of 2,000 pCi/ℓ is used.

TABLE 4.12-1
(Continued)

TABLE NOTATION

(1) This list does not mean that only these nuclides are to be considered. Other peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.3.

(2) Required detection capabilities for thermoluminescent dosimeters used for environmental measurements are given in Regulatory Guide 4.13.

(3) The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

$$LLD = \frac{(4.66)\ (s_b)}{(E)(V)(2.22)(Y)[\exp\ (-\lambda \Delta t)]}$$

Where:

LLD is the "a priori" lower limit of detection as defined above, as picocuries per unit mass or volume,

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate, as counts per minute,

E is the counting efficiency, as counts per disintegration,

V is the sample size in units of mass or volume,

2.22 is the number of disintegrations per minute per picocurie,

Y is the fractional radiochemical yield, when applicable,

λ is the radioactive decay constant for the particular radionuclide, and

Δt for environmental samples is the elapsed time between sample collection, or end of the sample collection period, and time of counting.

Typical values of E, V, Y and Δt should be used in the calculation.

TABLE 4.12-1
(Continued)

TABLE NOTATION

It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement. Analyses shall be performed in such a manner that the stated LLDs will be achieved under routine conditions. Occasionally background fluctuations, unavoidable small sample sizes, the presence of interfering nuclides, or other uncontrollable circumstances may render these LLDs unachievable. In such cases, the contributing factors shall be identified and described in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.3.

(4) LLD for drinking water samples. If no drinking water pathway exists, the LLD of gamma isotopic analysis may be used.

(5) An equilibrium mixture of the parent and daughter isotopes which corresponds to 15 pCi/ℓ of the parent isotope.

3/4.12.2 LAND USE CENSUS

LIMITING CONDITION FOR OPERATION

3.12.2 A Land Use Census shall be conducted and shall identify within a distance of 8 km (5 miles) the location in each of the 16 meteorological sectors of the nearest milk animal, the nearest residence, and the nearest garden* of greater than 50 m^2 (500 ft^2) producing broad leaf vegetation.

APPLICABILITY: At all times.

ACTION:

a. With a Land Use Census identifying a location(s) that yields a calculated dose or dose commitment greater than the values currently being calculated in Specification 4.11.2.3, pursuant to Specification 6.9.1.4, identify the new location(s) in the next Semiannual Radioactive Effluent Release Report

b. With a Land Use Census identifying a location(s) that yields a calculated dose or dose commitment (via the same exposure pathway) 20% greater than at a location from which samples are currently being obtained in accordance with Specification 3.12.1, add the new location(s) within 30 days to the Radiological Environmental Monitoring Program given in the ODCM. The sampling location(s), excluding the control station location, having the lowest calculated dose or dose commitment(s), via the same exposure pathway, may be deleted from this monitoring program after October 31 of the year in which this Land Use Census was conducted. Pursuant to Specification 6.14., submit in the next Semiannual Radioactive Effluent Release Report documentation for a change in the ODCM including a revised figure(s) and table(s) for the ODCM reflecting the new location(s) with information supporting the change in sampling locations.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.2 The Land Use Census shall be conducted during the growing season at least once per 12 months using that information that will provide the best results, such as by a door-to-door survey, aerial survey, or by consulting local agriculture authorities. The results of the Land Use Census shall be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.3.

* Broad leaf vegetation sampling may be performed at the SITE BOUNDARY in each of two different direction sectors with the highest predicted D/Qs in lieu of the garden census. Specifications for broad leaf vegetation sampling in Table 3.12-1, Part 4.b, shall be followed, including analysis of control samples.

3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.3 Analyses shall be performed on radioactive materials, supplied as part of an Interlaboratory Comparison Program that has been approved by the Commission.*

APPLICABILITY: At all times.

ACTION:

a. With analyses not being performed as required above, report the corrective actions taken to prevent a recurrence to the Commission in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.3.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.3 A summary of the results obtained as part of the above required Interlaboratory Comparison Program shall be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.3.

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* This condition is satisfied by participation in the Environmental Radioactivity Laboratory Intercomparison Studies Program conducted by the Environmental Protection Agency(EPA).

BASES FOR

SECTIONS 3.0 AND 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

NOTE

The BASES contained in succeeding pages summarize the reasons for the Specifications in Sections 3.0 are 4.0, but in accordance with 10 CFR 50.36 are not part of these Technical Specifications.

3/4.0 APPLICABILITY

BASES

The specifications of this section provide the general requirements applicable to each of the Limiting Conditions for Operation and Surveillance Requirements within Section 3/4. In the event of a disagreement between the requirements stated in these Technical Specifications and those stated in an applicable Federal Regulation or Act, the requirements stated in the applicable Federal Regulation or Act shall take precedence and shall be met.

3.0.1 This specification defines the applicability of each specification in terms of defined OPERATIONAL MODES or other specified conditions and is provided to delineate specifically when each specification is applicable.

3.0.2 This specification defines those conditions necessary to constitute compliance with the terms of an individual Limiting Condition for Operation and associated ACTION requirement.

3.0.3 The specification delineates the measures to be taken for those circumstances not directly provided for in the ACTION statements and whose occurrence would violate the intent of a specification. For example, Specification 3.5.2 requires four Safety Injection Pumps to be OPERABLE and provides explicit ACTION requirements if one or two pumps are inoperable. Under the requirements of Specification 3.0.3, if three or more of the required Safety Injection pumps are inoperable, within 1 hour measures must be initiated to place the unit in at least HOT STANDBY within the next 6 hours, and in at least HOT SHUTDOWN within the following 6 hours. As a further example, Specification 3.6.2.1 requires two Containment Spray Systems to be OPERABLE and provides explicit ACTION requirements if one Spray System is inoperable. Under the requirements of Specification 3.0.3 if both the required Containment Spray Systems are inoperable, within 1 hour measures must be initiated to place the unit in at least HOT STANDBY within the next 6 hours, in at least HOT SHUTDOWN within the following 6 hours, and in COLD SHUTDOWN within the subsequent 24 hours. It is acceptable to initiate and complete a reduction in OPERATIONAL MODES in a shorter time interval than required in the ACTION statement and to add the unused portion of this allowable out-of-service time to that provided for operation in subsequent lower OPERATIONAL MODE(S). Stated allowable out-of-service times are applicable regardless of the OPERATIONAL MODE(s) in which the inoperability is discovered but the times provided for achieving a mode reduction are not applicable if the inoperability is discovered in a mode lower than the applicable mode. For example if the Containment Spray Station was discovered to be inoperable while in STARTUP, the Action Statement would allow up to 156 hours to achieve COLD SHUTDOWN. If HOT STANDBY is attained in 16 hours rather than the allowed 78 hours, 140 hours would still be available before the plant would be required to be in COLD SHUTDOWN. However, if this system was discovered to be inoperable while in HOT STANDBY, the 6 hours provided to achieve HOT STANDBY would not be additive to the time available to achieve COLD SHUTDOWN so that the total allowable time is reduced from 156 hours to 150 hours.

3/4.0 APPLICABILITY

BASES

3.0.4 This specification provides that entry into an OPERATIONAL MODE or other specified applicability condition must be made with: (1) the full complement of required systems, equipment, or components OPERABLE and (2) all other parameters as specified in the Limiting Conditions for Operation being met without regard for allowable deviations and out-of-service provisions contained in the ACTION statements.

The intent of this provision is to ensure that facility operation is not initiated with either required equipment or systems inoperable or other specified limits being exceeded.

Exceptions to this provision have been provided for a limited number of specifications when startup with inoperable equipment would not affect plant safety. These exceptions are stated in the ACTION statements of the appropriate specifications.

3.0.5 This specification delineates what additional conditions must be satisfied to permit operation to continue, consistent with the ACTION statements for power sources, when a normal or emergency power source is not OPERABLE. It specifically prohibits operation when one division is inoperable because its normal or emergency power source is inoperable and a system, subsystem, train, component or device in another division is inoperable for another reason.

The provisions of this specification permit the ACTION statements associated with individual systems, subsystems, trains, components, or devices to be consistent with the ACTION statements of the associated electrical power source. It allows operation to be governed by the time limits of the ACTION statement associated with the Limiting Condition for Operation for the normal or emergency power source, not the individual ACTION statements for each system, subsystem, train, component or device that is determined to be inoperable solely because of the inoperability of its normal or emergency power source.

For example, Specification 3.8.1.1 requires in part that two emergency diesel generators be OPERABLE. The ACTION statement provides for an out-of-service time when one emergency diesel generator is not OPERABLE. If the definition of OPERABLE were applied without consideration of Specification 3.0.5, all systems, subsystems, trains, components and devices supplied by the inoperable emergency power source would also be inoperable. This would dictate invoking the applicable ACTION statements for each of the applicable Limiting Conditions for Operation. However, the provisions of specifications 3.0.5 permit the time limit for continued operation to be consistent with the ACTION statement for the inoperable emergency diesel generator instead, provided this would mean that the corresponding normal power source must be OPERABLE, and all redundant systems, subsystems, trains, components, and devices must be

3.0.5 (Continued)

OPERABLE, or otherwise satisfy Specification 3.0.5 (i.e., be capable of performing their design function and have at least one normal or one emergency power source OPERABLE). If they are not satisfied, shutdown is required in accordance with the specification.

In COLD SHUTDOWN or REFUELING condition Specification 3.0.5 is not applicable, and thus the individual ACTION statements for each applicable Limiting Condition for Operation in these conditions must be adhered to.

4.0.1 This specification provides that surveillance activities necessary to ensure the Limiting Conditions for Operation are met and will be performed during the OPERATIONAL MODES or other conditions for which the Limiting Conditions for Operation are applicable. Provisions for additional surveillance activities to be performed without regard to the applicable OPERATIONAL MODES or other conditions are provided in the individual Surveillance Requirements. Surveillance Requirements for Special Test Exceptions need only be performed when the Special Test Exception is being utilized as an exception to an individual specification.

4.0.2 The provisions of this specification provide allowable tolerances for performing surveillance activities beyond those specified in the nominal surveillance interval. These tolerances are necessary to provide operational flexibility because of scheduling and performance considerations. The phrase "at least" associated with a surveillance frequency does not negate this allowable tolerance value and permits the performance of more frequent surveillance activities.

The tolerance values, taken either individually or consecutively over three tests intervals, are sufficiently restrictive to ensure that the reliability associated with the surveillance activity is not significantly degraded beyond that obtained from the nominal specified interval.

4.0.3 The provisions of this specification set forth the criteria for determination of compliance with the OPERABILITY requirements of the Limiting Conditions for Operation. Under these criteria, equipment, systems or components are assumed to be OPERABLE if the associated surveillance activities have been satisfactorily performed within the specified time interval. Nothing in this provision is to be construed as defining equipment, systems or components OPERABLE when such items are found or known to be inoperable although still meeting the SURVEILLANCE REQUIREMENTS. Items may be determined inoperable during use, during surveillance tests, or in accordance with this specification. Therefore, ACTION statements are entered when the Surveillance Requirements should have been performed rather than at the time it is discovered that the tests were not performed.

4.0.4 This specification ensures that the surveillance activities associated with a Limiting Condition for Operation have been performed within the specified time interval prior to entry into an OPERATIONAL MODE or other applicable condition. The intent of this provision is to ensure that surveillance activities have been satisfactorily demonstrated on a current basis as required to meet the OPERABILITY requirements of the Limiting Condition for Operation.

Under the terms of this specification, for example, during initial plant STARTUP or following extended plant outages, the applicable surveillance activities must be performed within the stated surveillance interval prior to placing or returning the system or equipment into OPERABLE status.

4.0.5 This specification ensures that inservice inspection of ASME Code Class 1, 2 and 3 components and inservice testing of ASME Code Class 1, 2 and 3 pumps and valves will be performed in accordance with a periodically updated version of Section XI of the ASME Boiler and pressure Vessel Code and Addenda as required by 10 CFR 50.55a. Relief from any of the above requirements has been provided in writing by the Commission and is not a part of these Technical Specifications.

This specification includes a clarification of the frequencies for performing the inservice inspection and testing activities required by Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda. This clarification is provided to ensure consistency in surveillance intervals throughout these Technical Specifications and to remove any ambiguities relative to the frequencies for performing the required inservice inspection and testing activities.

Under the terms of this specification, the more restrictive requirements of the Technical Specifications take precedence over the ASME Boiler and Pressure Vessel Code and applicable Addenda. For example, the requirements of Specification 4.0.4 to perform surveillance activities prior to entry into an OPERATIONAL MODE or other specified applicability condition takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows pumps to be tested up to 1 week after return to normal operation. And for example, the Technical Specification definition of OPERABLE does not grant a grace period before a device that is not capable of performing its specified function is declared inoperable and takes precedence over the ASME Boiler and Pressure Vessel code provision which allows a valve to be incapable of performing its specified function for up to 24 hours before being declared inoperable.

3/4.1 REACTIVITY CONTROL SYSTEMS

BASES

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that: (1) the reactor can be made subcritical from all operating conditions, (2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and (3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. The maximum shutdown margin requirement occurs at end-of-core life and is based on the value used in analysis of the hypothetical steam break accident. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 1.77% $\Delta k/k$ is required to control the worst case reactivity transient. With T_{avg} less than 200 F, the reactivity transients resulting from a postulated steam line break cooldown are minimal and a 1% $\Delta k/k$ SHUTDOWN MARGIN provides adequate protection. Accordingly, the SHUTDOWN MARGIN requirements in these specifications are based on these limiting conditions and are consistent with FSAR safety analysis assumptions.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the value of this coefficient remains within the limiting condition assumed in the FSAR accident and transient analyses.

The MTC values of this specification are applicable to a specific set of plant conditions; accordingly, verification of MTC values at conditions other than those explicitly stated will require extrapolation to those conditions in order to permit an accurate comparison.

The most negative MTC, value equivalent to the most positive moderator density coefficient (MDC), was obtained by incrementally correcting the MDC used in the FSAR analyses to nominal operating conditions. These corrections involved subtracting the incremental change in the MDC associated with a core condition of all rods inserted (most positive MDC) to an all rods withdrawn condition and, a conversion for the rate of change of moderator density with temperature at RATED THERMAL POWER conditions. This value of the MDC was then

MODERATOR TEMPERATURE COEFFICIENT (Continued)

transformed into the limiting MTC value -3.5×10^{-4} Δk/k/ F. The MTC value of -3.0×10^{-4} Δk/k/ F in the Surveillance Requirements represents a conservative value (with corrections for burnup and soluble boron) at a core condition of 300 ppm equilibrium boron concentration and is obtained by making these corrections to the limiting MTC value of -3.5×10^{-4} Δk/k/ F at EOL.

The Surveillance Requirements for measurement of the MTC at the beginning and near the end of the fuel cycle are adequate to confirm that the MTC remains within its limits since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 541 F. This limitation is required to ensure: (1) the moderator temperature coefficient is within its analyzed temperature range, (2) the trip instrumentation is within its normal operating range, (3) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and (4) the reactor vessel is above its minimum RT_{NDT} temperature.

3/4.1.2 BORATION SYSTEMS

The Boron Injection System ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include: (1) borated water sources, (2) charging pumps, (3) separate flow paths, (4) boric acid transfer pumps, (5) associated Heat Tracing Systems, and (6) an emergency power supply from OPERABLE diesel generators for vital system components.

With the RCS average temperature above 200 F, a minimum of two boron injection flow paths are required to ensure single functional capability in the event an assumed failure renders one of the flow paths inoperable. The boration capability of either flow path is sufficient to provide a SHUTDOWN MARGIN from expected operating conditions of 1.77% Δk/k after xenon decay and cooldown to 200 F.

3/4.1.2 BORATION SYSTEMS (Continued)

With the RCS temperature below 200 F, one Boron Injection System is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity changes in the event the single Boron Injection System becomes inoperable.

The 45 gpm flow verification of a charging pump in Modes 5 and 6 is to verify that the charging pump is capable of delivering flow in the shutdown mode. The 45 gpm has no accident analysis basis. The test of the charging pumps in accordance with Specification 4.0.5 is conducted in Modes 3 or 4 when the RCS is pressurized in order to establish a meaningful test on the positive displacement charging pumps.

The OPERABILITY of one Boron Injection System during REFUELING ensures that this system is available for reactivity control while in MODE 6.

The OPERABILITY of the redundant heat tracing channels associated with the boric acid tank system ensures that the solubility of the boron solution will be maintained above the solubility limit of 135 F at 22,500 ppm boron.

One channel of heat tracing is sufficient to maintain the specified temperature limit. Since one channel of heat tracing is sufficient to maintain the specified temperature, operation with one channel out-of-service is permitted for a period of 30 days provided additional temperature surveillance is performed.

3/4 .1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that: (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) limit the potential effects of rod misalignment on associated accident analyses. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits. OPERABLE condition for the analog rod position indicators is defined as being capable of indicating rod position to within ± 12 steps of the REFERENCE POSITION. For the Shutdown Banks and Control Banks A and B, the REFERENCE POSITION is defined as the group demand counter indicated position between 0 and 30 steps withdrawn inclusive, and between 200 and 228 steps withdrawn inclusive. This permits the operator to verify that the control rods in these banks are either fully withdrawn or fully inserted, the normal operating modes for these banks. Knowledge of these bank positions in these two areas satisfies all accident analysis assumptions concerning their position. For Control Banks C and D the REFERENCE POSITION is defined as the group demand counter indicated positions between 0 and 30 steps withdrawn inclusive, and between 150 and 228 steps withdrawn inclusive. For the withdrawal range of 31 to 149 steps inclusive the REFERENCE POSITION is defined as the individual rod calibration

MOVABLE CONTROL ASSEMBLIES (Continued)

curve noting indicated analog rod position vs indicated group demand counter position. Comparison of the indicated analog rod position to the calibration curve is sufficient to allow determination that a control rod is indeed misaligned from its bank. Comparison of the group demand counters to the bank insertion limits with verification of rod position with the analog rod position indicators (after thermal soak after rod motion) is sufficient verification that the control rods are above the insertion limits.

Rod position indication is provided by two methods: a digital count of actuating pulses which shows demand position of the banks and a linear position indicator Linear Variable Differential Transformer which indicates the actual rod position. The relative accuracy of the linear position indicator Linear Variable Differential Transformer is such that, with the most adverse error, an alarm will be actuated if any two rods within a bank deviate by more than 24 steps for rods in motion and 12 steps for rods at rest. Complete rod misalignment (12 feet out of alignment with its bank) does not result in exceeding core limits in steady-state operation at RATED THERMAL POWER.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met. Misalignment of a rod requires measurement of peaking factors and a restriction in THERMAL POWER. These restrictions provide assurance of fuel rod integrity during continued operation. In addition, those safety analyses affected by a misaligned rod are re-evaluated to confirm that the results remain valid during future operation.

The maximum rod drop time restriction is consistent with the assumed rod drop time in the safety analyses. Measurement with T_{avg} greater than or equal to 541 F and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a Reactor trip at operating conditions.

Control rod positions and OPERABILITY of the rod position indicators are required to be verified on a nominal basis of once per 12 hours with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCOs are satisifed.

BASES

3/4.2.1 AXIAL FLUX DIFFERENCE

The limits on AXIAL FLUX DIFFERENCE (AFD) assure that the $F_Q(Z)$ upper bound envelope of 2.32 times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

Target flux difference is determined at equilibrium xenon conditions. The full-length rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady-state operation at high power levels. The value of the target flux difference obtained under these conditions divided by the fraction of RATED THERMAL POWER is the target flux difference at RATED THERMAL POWER for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RATED THERMAL POWER value by the appropriate fractional THERMAL POWER level. The periodic updating of the target flux difference value is necessary to reflect core burnup considerations.

Strict control of the flux difference (and rod position) is not as necessary during operation at less than 90% power. This is because xenon distribution control is not as significant as the control at full power and allowance has been made in predicting the heat flux peaking factors for less strict control at less than 90% power. Although it is intended that the Plant will be operated with the AFD within the target band required by Specification 3.2.1 about the target flux difference, during rapid plant THERMAL POWER reductions, control rod motion will cause the AFD to deviate outside of the target band at reduced THERMAL POWER levels. This deviation will not affect the xenon redistribution sufficiently to change the envelope of peaking factors which may be reached on a subsequent return to RATED THERMAL POWER (with the AFD within the target band) provided the time duration of the deviation is limited. Accordingly, a 1-hour penalty deviation limit cumulative during the previous 24 hours is provided for operation outside of the target band but within the limits of Figure 3.2-1 while at THERMAL POWER levels between 50% and 90% of RATED THERMAL POWER. If the flux difference exceeds the limit for a cumulative period of one hour in any 24 hours, then xenon distributions may be significantly changed and hence operation at less than 50% power is required to protect against potentially more severe consequences of some accidents. For THERMAL POWER levels below 50% of RATED THERMAL POWER, deviations of the AFD outside of the target band are less significant. The penalty of 2 hours actual time reflects this reduced significance.

Strict control of flux differences is not possible during certain physics tests or during the required periodic excore calibrations. Therefore, this specification is not applicable during physics testing or excore calibration provided the duration of the deviation is limited. This is acceptable due to the extremely low probability of a significant accident during these operations.

AXIAL FLUX DIFFERENCE (Continued)

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD Monitor Alarm. The computer determines the 1-minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for two or more OPERABLE excore channels are outside the target band and the THERMAL POWER is greater than 90% of RATED THERMAL POWER. During operation at THERMAL POWER levels between 50% and 90% and below 50% RATED THERMAL, the computer outputs an alarm message when the penalty deviation accumulates beyond the limits of 1 hour and 2 hours, respectively.

Figure B 3/4 2-1 shows a typical monthly target band.

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR, AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

The limits on heat flux hot channel factor and nuclear enthalpy rise hot channel factor ensure that: (1) the design limits on peak local power density and minimum DNBR are not exceeded and (2) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200 degrees-F ECCS acceptance criteria limit. The LOCA peak fuel clad temperature limit may be sensitive to the number of steam generator tubes plugged. The current limit is valid for tube plugging levels up to 5%.

$F_Q(Z)$, Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux.

$F^N_{\Delta H}$, Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

Each of these is measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to ensure that the limits are maintained provided:

a. Control rods in a single group move together with no individual rod insertion differing by more than ± 12 steps, indicated, from the group demand position;

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.6.

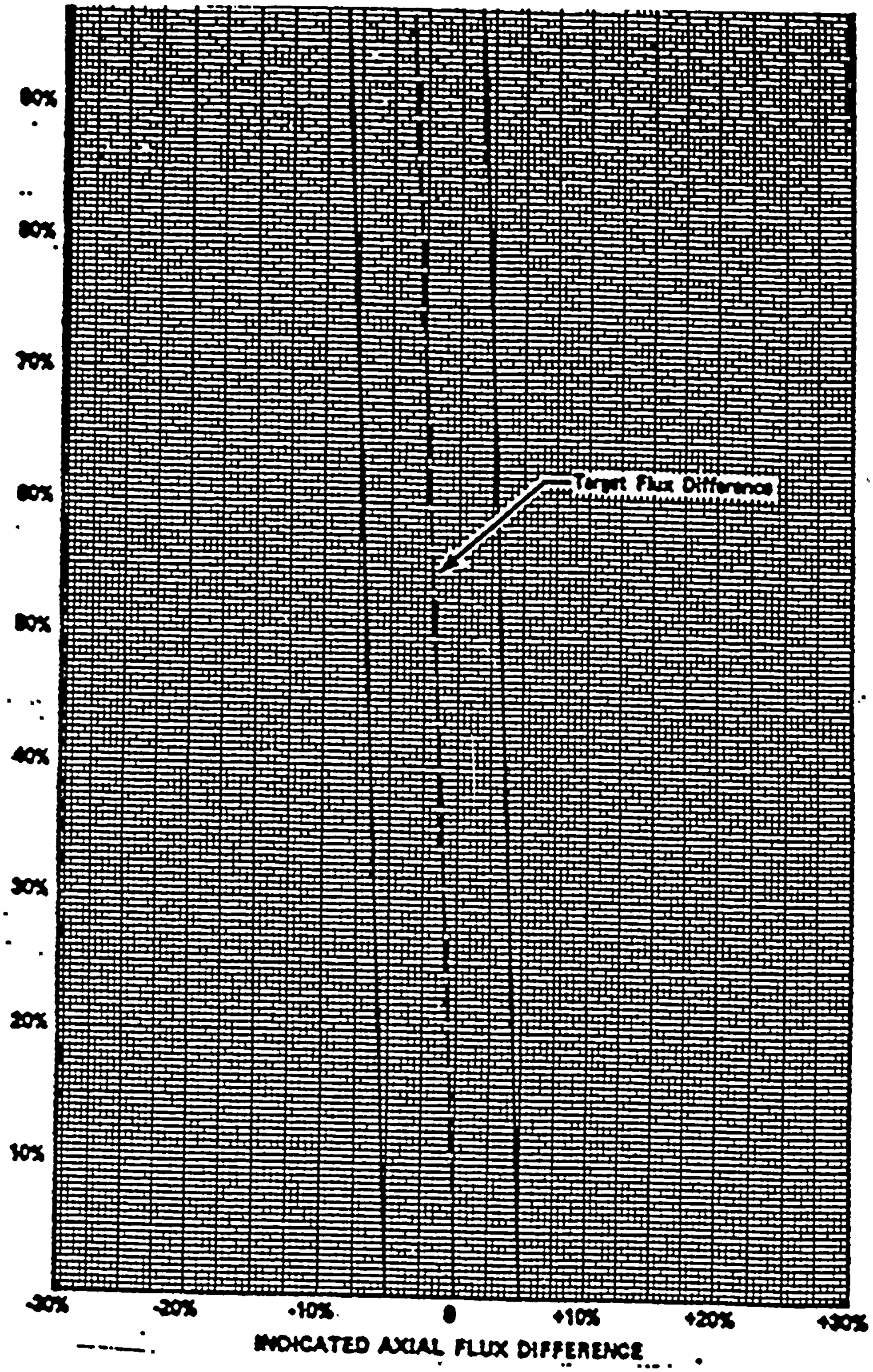


FIGURE B 3/4 2-1

TYPICAL INDICATED AXIAL FLUX DIFFERENCE VERSUS THERMAL POWER

HEAT FLUX HOT CHANNEL FACTOR, AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

c. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained; and

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

When an F_Q measurement is taken, both experimental error and manufacturing tolerance must be allowed for. Five percent is the appropriate allowance for a full core map taken with the movable incore detector flux mapping system and three percent is the appropriate allowance for manufacturing tolerance. These uncertainties only apply if the map is taken for purposes other than the determination of P_{BL} and P_{RB}.

$F^N_{\Delta H}$ will be maintained within its limits provided Conditions a. through d. above are maintained.

In the specified limit of $F^N_{\Delta H}$, there is an 8 percent allowance for uncertainties which means that normal operation of the core is expected to result in $F^N_{\Delta H} \leq 1.62/1.08$. The logic behind the larger uncertainty in this case is that (a) normal perturbations in the radial power shape (e.g., rod misalignment) affect $F^N_{\Delta H}$, in most cases without necessarily affecting F_Q, (b) although the operator has a direct influence on F_Q through movement of rods, and can limit it to the desired value, he has no direct control over $F^N_{\Delta H}$ and (c) an error in the prediction for radial power shape, which may be detected during startup physics tests can be compensated for in F_Q by tighter axial control, but compensation for $F_{\Delta H}$ is less readily available. When a measurement of $F^N_{\Delta H}$ is taken, experimental error must be allowed for and 4% is the appropriate allowance for a full core map taken with the movable incore detector flux mapping system.

HEAT FLUX HOT CHANNEL FACTOR, AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

The analytically determined $[F_Q]^P$ is formulated to generate limiting shapes for all load follow maneuvers consistent with control to a ± 5% band about the target flux difference. For Base Load operation the severity of the shapes that need to be considered is significantly reduced relative to load follow operation.

The severity of possible shapes is small due to the restrictions imposed by Sections 4.2.2.3. To quantify the effect of the limiting transients which could occur during Base Load operation, the function $W(Z)_{BL}$ is calculated from the following relationship:

$$W(Z)_{BL} = \text{Max}\left[\frac{F_Q(Z)(\text{Base Load Case(s)}, 150\text{ MWD/T})}{F_Q(Z)\ (\text{ARO}, 150\text{ MWD/T})}, \frac{F_Q(Z)(\text{Base Case(s)}, 85\%\text{ EOL BU})}{F_Q(Z)\ (\text{ARO}, 85\%\text{ BOL BU})}\right]$$

For Radial Burndown operation the full spectrum of possible shapes consistent with control to a ± 5% Delta-I band needs to be considered in determining power capability. Accordingly, to quantify the effect of the limiting transients which could occur during Radial Burndown operation, the function $F_z(Z)$ is calculated from the following relationship:

$$F_z(Z) = [F_Q(Z)]\ \text{FAC Analysis}/[F_{xy}(Z)]\ \text{ARO}$$

The essence of the procedure is to maintain the xenon distribution in the core as close to the equilibrium full power condition as possible. This can be accomplished by using the boron system to position the full length control rods to produce the required indicated flux difference.

Above the power level of P_T, additional flux shape monitoring is required. In order to assure that the total power peaking factor, F_Q, is maintained at or below the limiting value, the movable incore instrumentation will be utilized. Thimbles are selected initially during startup physics tests so that the measurements are representative of the peak core power density. By limiting the core average axial power distribution, the total power peaking factor F_Q can be limited since all other components remain relatively fixed. The remaining part of the total power peaking factor can be derived

from incore measurements, i.e., an effective radial peaking factor $\bar{R}$, can be determined as the ratio of the total peaking factor resulting from a full core flux map and the axial peaking factor in a selected thimble.

HEAT FLUX HOT CHANNEL FACTOR, AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

The limiting value of $[F_j (Z)]_S$ is derived as follows:

$$[F_j (Z)]_S = \frac{[F_Q]^L \times [K(Z)]}{P_L \bar{R}_j (1+ \sigma_j) (1.03)(1.07)}$$

Where:

a) F_j (Z) is the normalized axial power distribution from thimble j at elevation Z.

b) P_L is reactor thermal power expressed as a fraction of 1.

c) K (Z) is the reduction in the F_Q limit as a function of core elevation (Z) as determined from Figure 3.2-2.

d) $[F_j (Z)]_S$ is the alarm setpoint for MIDS.

e) $\bar{R}_j$, for thimble j, is determined from n=6 incore flux maps covering the full configuration of permissible rod patterns at the thermal power limit of P_T.

$$\bar{R}_j = \frac{\sum_{i=1}^{n} R_{ij}}{n}$$

where

$$R_{ij} = \frac{F_{Qi} \text{ meas.}}{[F_{ij} (Z)] \text{ max}}$$

and F_{ij} (Z) is the normalized axial distribution at elevation Z from thimble j in map i which has a measured peaking factor without uncertainties or densification allowance of F_{Qi} meas.

f) σ_j is the standard deviation, expressed as a fraction or percentage of $\bar{R}_j$, and is derived from n flux maps and the relationship below, or 0.02 (2%), whichever is greater.

$$\sigma_j = \frac{\left[\frac{1}{n-1} \sum_{i=1}^{n} (R_{ij} - \bar{R}_j)^2 \right]^{1/2}}{\bar{R}_j}$$

HEAT FLUX HOT CHANNEL FACTOR, AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

g) The factor 1.03 reduction in the kw/ft limit is the engineering uncertainty factor.

h) The factors $(1+\sigma_j)$ and 1.07 represent the margin between $(F_j(Z)]_L$ limit and the MIDS alarm setpoint $[F_j(Z)]_S$. Since $(1+\sigma_j)$ is bounded by a lower limit of 1.02, there is at least a 9% reduction of the alarm setpoint. Operations are permitted in excess of the operational limit $\leq$ 4% while making power adjustment on a percent for percent basis.

3/4.2.4 QUADRANT POWER TILT RATIO

The QUADRANT POWER TILT RATIO limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during STARTUP testing and periodically during power operation.

The limit of 1.02, at which corrective action is required, provides DNB and linear heat generation rate protection with x-y plane power tilts. A limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

The 2-hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned control rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the maximum allowed power by 3% for each percent of tilt in excess of 1.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the moveable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the QUADRANT POWER TILT RATIO.

3/4.2.5 DNB PARAMETERS

The limits on the DNB-related parameters assure that each of the parameters are maintained within the normal steady-state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR above the applicable design limit throughout each analyzed transient. The indicated T_{avg} value of 576.3 F and the indicated pressurizer pressure value of 2237 psia correspond to analytical limits of 578.2 F and 2220 psia respectively, with the allowance for measurement uncertainty.

POWER DISTRIBUTION LIMITS

BASES

DNB PARAMETERS (Continued)

The indicated RCS flow value of 277,900 gpm corresponds to an analytical limit of 268,500 gpm with an allowance for measurement uncertainty.

The 12-hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation.

The above measurement uncertainty estimates assume that these instrument channel outputs are averaged to minimize the uncertainty.

3/4.3.1 and 3/4.3.2 REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

The OPERABILITY of the Reactor Trip System and the Engineered Safety Features Actuation System instrumentation and interlocks ensures that: (1) the associated ACTION and/or Reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its Setpoint, (2) the specified coincidence logic is maintained, (3) redundancy is maintained as much as possible to permit a channel to be out-of-service for testing or maintenance, and (4) sufficient system functional capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy, and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the safety analyses. The Surveillance Requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

The engineered Safety Features Actuation System Instrumentation Trip Setpoints specified are the nominal values at which the trips are set for each functional unit. A setpoint is considered to be adjusted consistent with the nominal value when the "as measured" Setpoint is within the band allowed for calibration accuracy.

To accommodate the instrument drift assumed to occur between operational tests and the accuracy to which Setpoints can be measured and calibrated, Allowable Values for the Setpoints have been specified. Operation with Setpoints less conservative than the Trip Setpoint but within the Allowable Value is acceptable since an allowance has been made in the safety analysis to accommodate this error.

The methodology to derive the Trip Setpoints is based upon combining all of the uncertainties in the channels. Inherent to the determination of the Trip Setpoints are the magnitudes of these channel uncertainties. Sensor and rack instrumentation utilized in these channels are expected to be capable of operating within the allowances of these uncertainty magnitudes. Rack drift in excess of the Allowable Value exhibits the behavior that the rack has not met its allowance. Being that there is a small statistical chance that this will happen, an infrequent excessive drift is expected. Rack or sensor drift, in excess of the allowance that is more than occasional, may be indicative of more serious problems and should warrant further investigation.

3/4.3.1 and 3/4.3.2 REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

3/4.3.1 CONTINUED)

BASES

The measurement of response time at the specified frequencies provides assurance that the Reactor Trip and the Engineered Safety Features actuation associated with each channel is completed within the time limit assumed in the safety analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable. Response time may be demonstrated by any series of sequential, overlapping, or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either: (1) in place, onsite, or offsite test measurements, or (2) utilizing replacement sensors with certified response time.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING FOR PLANT OPERATIONS

The OPERABILITY of the radiation monitoring instrumentation for plant operations ensures that conditions indicative of potential uncontrolled radioactive releases are monitored and that appropriate actions will be automatically or manually initiated when the radiation level monitored by each channel reaches its alarm or trip setpoint.

3/4.3.3.2 MOVABLE INCORE DETECTOR

The OPERABILITY of the movable incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the core. The OPERABILITY of this system is demonstrated by irradiating each detector used and determining the acceptability of its voltage curve.

For the purpose of measuring $F_Q(Z)$ or $F^N_{\Delta H}$ a full incore flux map is used.

3/4.3.3.3 SEISMIC INSTRUMENTATION

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the facility to determine if plant shutdown is required pursuant to Appendix A of 10 CFR Part 100.

3/4.3.3.4 METEOROLOGICAL INSTRUMENTATION

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data are available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public and is consistent with the recommendations of Regulatory Guide 1.23, "Onsite Meteorological Programs," February 1972.

3/4.3.3.5 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, Revision 3, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Folloiwng an Accident," May 1983 and NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

3/4.3.3.6 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The Alarm/Trip Setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR.

3/4.3.3.7 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

The Radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The Alarm/Trip Setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. This instrumentation also includes provisions for monitoring (and controlling) the concentrations of potentially explosive gas mixtures in the GAS DECAY TANK SYSTEM. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

3/4.3.4 TURBINE OVERSPEED PROTECTION

This specification is provided to ensure that the turbine overspeed protection instrumentation and the turbine speed control valves are OPERABLE and will protect the turbine from excessive overspeed. Protection from turbine excessive overspeed is required since excessive overspeed of the turbine could generate potentially damaging missiles which could impact and damage safety-related components, equipment or structures.

3/4.4 REACTOR COOLANT SYSTEM

BASES

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with all reactor coolant loops in operation and maintain DNBR above the applicable design limit during all normal operations and anticipated transients. In MODES 1 and 2 with one reactor coolant loop not in operation this specification requires that the plant be in at least HOT STANDBY within 6 hours.

In MODE 3, two reactor coolant loops provide sufficient heat removal capability for removing core decay heat even in the event of a bank withdrawal accident; however, a single reactor coolant loop provides sufficient heat removal capacity if a bank withdrawal accident can be prevented, i.e., by opening the Reactor Trip System breakers. Single failure considerations require that two loops be OPERABLE at all times.

In MODE 4, and in MODE 5 with reactor coolant loops filled, a single reactor coolant loop or RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops (either RHR or RCS) be OPERABLE.

In MODE 5 with reactor Coolant loops not filled, a single RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations, and the unavailability of the steam generators as a heat removing component, require that at least two RHR loops be OPERABLE.

The operation of one reactor coolant pump (RCP) or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

The restrictions on starting a RCP with one or more RCS cold legs less than or equal to 275 F are provided to prevent RCS pressure transients, caused by energy additions from the Secondary Coolant System, which could exceed the limits of Appendix G to 10CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by either: (1) restricting the water volume in the pressurizer and thereby providing a volume for the reactor coolant to expand into, or (2) by restricting the starting of the RCPs to when the secondary water temperature of each steam generator is less than 50 F above each of the RCS cold leg temperatures.

The Technical Specifications for Hot Shutdown and Cold Shutdown allow an inoperable RHR pump to be the operating RHR pump for up to 2 hours for surveillance testing to establish operability. This is required because of the piping arrangement when the RHR system is being used for Decay Heat Removal.

3/4.4.2 SAFETY VALVES

The pressurizer Code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 288,000 lbs. per hour of saturated steam at the valve setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an RCS vent opening of at least 2.20 square inches will provide overpressure relief capability and will prevent RCS overpressurization. In addition, the Overpressure Mitigating System provides a diverse means of protection against RCS overpressurization at low temperatures.

During operation, all pressurizer Code safety valves must be OPERABLE to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss-of-load assuming no Reactor trip until the first Reactor Trip system Trip Setpoint is reached (i.e., no credit is taken for a direct Reactor trip on the loss-of-load) and also assuming no operation of the power-operated relief valves or steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Vessel Code.

3/4.4.3 PRESSURIZER

The limit on the maximum water volume in the pressurizer assures that the parameter is maintained within the normal steady-state envelope of operation assumed in the FSAR. The limit is consistent with the initial FSAR assumptions. The 12-hour periodic surveillance is sufficient to ensure that the parameter is restored to within its limit following expected transient operation. The maximum water volume also ensures that a steam bubble is formed and thus the RCS is not a hydraulically solid system. The requirement that the pressurizer heaters be OPERABLE provides capability of the plant to control Reactor Coolant System pressure and establish natural circulation.

3/4.4.4 PORV BLOCK VALVES

The power-operated relief valves (PORVs) and steam bubble function to relieve RCS pressure during all design transients up to and including the design step load decrease with steam dump. Operation of the PORVs minimizes the undesirable opening of the spring-loaded pressurizer code safety valves. The opening of the PORVs fulfills no safety-related function and no credit is taken for their operation in the safety analysis for MODE 1, 2 or 3.

Each PORV has a remotely operated block valve to provide a positive shutoff capability should a relief valve become inoperable.

3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the Reactor Coolant System and the Secondary Coolant System (primary-to-secondary leakage = 500 gallons per day per steam generator). Cracks having a reactor-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of 500 gallons per day steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required for all tubes with imperfections exceeding the plugging limit of 40% of the tube nominal wall thickness. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be promptly reported to the Commission per 10CFR50.72 and in a Special Report pursuant to Specification 6.9.2 within 30 days and prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS Leakage Detection Systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary to the containment. The containment sump level system is the normal sump level instrumentation. The Post Accident Containment Water Level Monitor - Narrow range instrumentation also satisfies the requirement for a sump level monitoring system. These Detection Systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection System," May 1973. In addition, gross leakage will be detected by changes in makeup water requirements, visual inspection, and audible detection. Leakage to other systems will be detected by activity changes (e.g., within the component cooling systems) or water inventory changes (e.g., tank levels).

3/4.4.6.2 OPERATIONAL LEAKAGE

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 gpm. This threshold value is sufficiently low to ensure early detection of additional leakage. When UNIDENTIFIED LEAKAGE is detected, an evaluation will be performed to identify the source. Any UNIDENTIFIED LEAKAGE shall not be interpreted as PRESSURE BOUNDARY LEAKAGE.

The total steam generator tube leakage limit of 1 gpm for all steam generators ensures that the dosage contribution from the tube leakage will be limited to a small fraction of 10 CFR Part 100 dose guideline values in the event of either a steam generator tube rupture or steam line break. The 1 gpm limit is consistent with the assumptions used in the analysis of these accidents. The 500 gpd leakage limit per steam generator ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions.

The 10 gpm IDENTIFIED LEAKAGE limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the Leakage Detection Systems.

OPERATIONAL LEAKAGE (Continued)

The 1 gpm leakage from any RCS pressure isolation valve is sufficiently low to ensure early detection of possible in-series check valve failure. It is apparent that when pressure isolation is provided by two in-series check valves and when failure of one valve in the pair can go undetected for a substantial length of time, verification of valve integrity is required. Since these valves are important in preventing overpressurization and rupture of the ECCS low pressure piping which could result in a LOCA that bypasses containment, these valves should be tested periodically to ensure low probability of gross failure.

The Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA.

3/4.4.7 CHEMISTRY

The limitations on Reactor Coolant System ensure that corrosion of the Reactor System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady-State limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride, and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady-State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady-State Limits.

The Surveillance Requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the reactor coolant ensure that the resulting 2-hour doses at the SITE BOUNDARY will not exceed an appropriately small fraction of 10 CFR Part 100 dose guideline values following a steam generator tube rupture accident in conjunction with an

SPECIFIC ACTIVITY (Continued)

assumed steady-state reactor-to-secondary steam generator leakage rate of 1 gpm. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the Turkey Point site, Units 3 and 4, such as SITE BOUNDARY location and meteorological conditions, were not considered in this evaluation.

The ACTION statement permitting power operation to continue for limited time periods with the reactor coolant's specific acitvity greater than 1 microCurie/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER. Operation with specific activity levels exceeding 1 microCurie/gram DOSE EQUIVALENT I-131 but within the limits shown on Figure 3.4-1 must be restricted since the activity levels allowed by Figure 3.4-1 increase the 2-hour thyroid dose at the SITE BOUNDARY by a factor of up to 20 following a postulated steam generator tube rupture.

The sample analysis for determining the gross specific activity and $\bar{E}$ can exclude the radioiodines because of the low reactor coolant limit of 1 microcurie/gram DOSE EQUIVALENT I-131, and because, if the limit is exceeded, the radioiodine level is to be determined every 4 hours. If the gross specific activity level and radioiodine level in the reactor coolant were at their limits, the radioiodine contribution would be approximately 1%. In a release of reactor coolant with a typical mixture of radioactivity, the actual radioiodine contribution would probably be about 20%. The exclusion of radio-nuclides with half-lives less than 30 minutes from these determinations has been made for several reasons. The first consideration is the difficulty to identify short-lived radionuclides in a sample that requires a significant time to collect, transport, and analyze. The second consideration is the predictable delay time between the postulated release of radioactivity from the reactor coolant to its release to the environment and transport to the SITE BOUNDARY.

For these reasons the radionuclides that are excluded from consideration are expected to decay to very low levels before they could be transported from the reactor coolant to the SITE BOUNDARY under any accident condition.

Reducing T_{avg} to less than 500 F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the reactor coolant is below the lift pressure of the atmospheric steam relief valves. The Surveillance Requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to take corrective action. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

All components in the RCS are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. During RCS heatup and cooldown, the temperature and pressure changes must be limited to be consistent with design assumptions and to satisfy stress limits for brittle fracture.

The temperature and pressure changes during heatup and cooldown are limited to be consistent with the following requirements as given in the ASME Boiler Pressure Vessel Code, Section III, Appendix G:

1. The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figures 3.4-2 to 3.4-5 for the service period specified thereon:

   a. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation; and

   b. Figures 3.4-2 to 3.4-5 define limits to assure prevention of non-ductile failure only. For normal operation, other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

2. These limit lines shall be calculated periodically using methods provided below,

3. The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the steam generator is below 70 F,

4. The pressurizer heatup and cooldown rates shall not exceed 100 F/h and 200 F/h, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320 F, and

5. System preservice hydrotests and inservice leak and hydrotests shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT}, at the end of 10 effective full power years (EFPY) of service life. The 10 EFPY service life period is chosen such that the limiting RT_{NDT} at the 1/4T location in the core region is greater than the RT_{NDT} of the limiting unirradiated material. The selection of such a limiting RT_{NDT} assures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

The reactor vessel materials have been tested to determine their initial RT_{NDT}. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation can cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence and copper content of the material in question, has been predicted. The heatup and cooldown limit curves of Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{NDT} at the end of 10 EFPY as well as adjustments for possible errors in the pressure and temperature sensing instruments.

The predicted shift in RT_{NDT} (ΔRT_{NDT}) determined in this manner may be used until the results from the material surveillance program, evaluated according to ASTM E185, are available. Capsules will be removed in accordance with the requirements of ASTM E185-73 and 10 CFR Part 50, Appendix H. The surveillance specimen withdrawal schedule is shown in Table 4.4-5. The results obtained from the surveillance specimens can be used to predict future radiation damage to the reactor vessel material. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule exceeds the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures, a semielliptical surface defect with a depth of one-quarter of the wall thickness, T, and a length of 3/2T is assumed to exist at the inside of the vessel wall as well as at the outside of the vessel wall. The dimensions of this postulated crack, referred to in Appendix G of ASME Section III as the reference flaw, amply exceed the current capabilities of inservice inspection techniques. Therefore, the reactor operation limit curves developed for this reference crack are conservative and provide sufficient safety margins for protection against nonductile failure.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

To assure that the radiation embrittlement effects are accounted for in the calculation of the limit curves, the most limiting value of the nil-ductility reference temperature, RT_{NDT}, is used and this includes the radiation-induced shift, ΔRT_{NDT}, corresponding to the end of the period for which heatup and cooldown curves are generated.

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{IR}, for the metal temperature at that time. K_{IR} is obtained from the reference fracture toughness curve, defined in Appendix G to the ASME Code.

At any time during the heatup or cooldown transient, K_{IR} is determined by the metal temperature at the tip of the postulated flaw, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from temperature gradients through the vessel wall are calculated and then the corresponding thermal stress intensity factor, K_{IT}, for the reference flaw is computed. From the calculated thermal stress intensity factors, the allowable pressures are determined based on the requirement that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time cannot exceed K_{IR}.

COOLDOWN

For the calculation of the allowable pressure versus coolant temperature during cooldown, the Code reference flaw is assumed to exist at the inside of the vessel wall. During cooldown, the controlling location of the flaw is always at the inside of the wall because the thermal gradients produce tensile stresses at the inside, which increases with increasing cooldown rates. Allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations. From these relations, composite limit curves are constructed for each cooldown rate of interest.

The use of the composite curve in the cooldown analysis is necessary because control of the cooldown procedure is based on measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed flaw. During cooldown, the 1/4T vessel location is at a higher temperature than the fluid adjacent to

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

the vessel ID. This condition, of course, is not true for the steady-state situation. It follows that at any given reactor coolant temperature, the delta T developed during cooldown results in a higher value of K_{IR}

at the 1/4T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist such that the increase in K_{IR} exceeds K_{IT}, the calculated allowable pressure during cooldown will be greater than the steady-state value.

The above procedures are needed because there is no direct control on temperature at the 1/4T location; therefore, allowable pressures may unknowingly be violated if the rate of cooling is decreased at various intervals along a cooldown ramp. The use of the composite curve eliminates this problem and assures conservative operation of the system for the entire cooldown period.

HEATUP

Three separate calculations are required to determine the limit curves for finite heatup rates. As is done in the cooldown analysis, allowable pressure-temperature relationships are developed for steady-state conditions as well as finite heatup rate conditions assuming the presence of a 1/4T defect at the inside of the vessel wall. The thermal gradients during heatup produce compressive stresses at the inside of the wall that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature; therefore, the K_{IR} for the 1/4T crack during heatup is lower than the K_{IR} for the 1/4T crack during steady-state conditions at the same coolant temperature. During heatup, especially at the end of the transient, conditions may exist such that the effects of compressive thermal stresses and different K_{IR}'s for steady-state and finite heatup rates do not offset each other and the pressure-temperature curve based on steady-state conditions no longer represents a lower bound of all similar curves for finite heatup rates when the 1/4T flaw is considered. Therefore, both cases have to be analyzed in order to assure that at any coolant temperature the lower value of the allowable pressure calculated for steady-state and finite heatup rates is obtained.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

The second portion of the heatup analysis concerns the calculation of pressure-temperature limitations for the case in which a 1/4T deep outside surface flaw is assumed. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and thus tend to reinforce any pressure stresses present. These thermal stresses, of course, are dependent on both the rate of heatup and the time (or coolant temperature) along the heatup ramp. Furthermore, since the thermal stresses at the outside are tensile and increase with increasing heatup rate, a lower bound curve cannot be defined. Rather, each heatup rate of interest must be analyzed on an individual basis.

Following the generation of pressure-temperature curves for both the steady-state and finite heatup rate situations, the final limit curves are produced as follows. A composite curve is constructed based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the least of the three values taken from the curves under consideration.

The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling condition switches from the inside to the outside and the pressure limit must at all times be based on analysis of the most critical criterion.

Finally, the new 10CFR50 Appendix G rule which addresses the metal temperature of the closure head flange and vessel flange regions is considered. The rule states that the minimum metal temperature for the flange regions should be at least 120 F higher than the limiting RT_{NDT} for these regions when the pressure exceeds 20 percent of the preservice hydrostatic test pressure (621 psig). Since the limiting RT_{NDT} for the flange regions for Turkey Point Units 3 and 4 is 44 F, the minimum temperature required for pressure of 621 psig and greater based on the Appendix G rule is 164 F. The Heatup and Cooldown curves as shown in Figures 3.4-2 to 3.4-5 clearly satisfy the above requirement by ample margins.

Although the pressurizer operates in temperature ranges above those for which there is reason for concern of nonductile failure, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

LOW TEMPERATURE OVERPRESSURE PROTECTION

The Technical Specifications provide requirements to isolate High Pressure Safety Injection from the RCS by specifying the closure of MOV-843A, 843B, and 869 and the removal of their power supplies, and to prevent the start of an idle RCP if secondary temperature is more than 50 F above the RCS cold leg temperatures. These requirements are designed to ensure that mass and heat input transients more severe than those assumed in the low temperature overpressurization protection analysis cannot occur.

The OPERABILITY of two PORVs or an RCS vent opening of at least 2.20 square inches ensures that the RCS will be protected from analyzed pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 275 F. Either PORV has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either: (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 50 F above the RCS cold leg temperatures (including margin for instrument error) or (2) the start of a HPSI pump and its injection into a water-solid RCS.

REACTOR MATERIAL SURVEILLANCE PROGRAM

Each Type I capsule contains 28 Charpy V-notch specimens, ten Charpy specimens machined from each of the two shell forgings. The remaining eight Charpy specimens are machined from correlated monitor material. In addition, each Type I capsule contains four tensile specimens (two specimens from each of the two shell forgings). Dosimeters of copper, nickel, aluminum-cobalt, and cadmium-shielded aluminum-cobalt wire are secured in holes drilled in spacers at the top, middle and bottom of each Type I capsule.

Each Type II capsule contains 32 Charpy V-notch specimens: eight specimens machined from one of the shell forgings, eight specimens of weld metal and eight specimens of HAZ metal, the remaining eight specimens are correlation monitors. In addition, each Type II capsule contains four tensile specimens and four WOL forgings and the weld metal. Each Type II capsule contains a dosimeter block at the center of the capsule. Two cadmium-oxide-shielded capsules, containing the two isotopes uranium-238 and neptunium-237, are contained in the dosimeter block. The double containment afforded by the dosimeter assembly prevents loss and contamination by the neptunium-237 and uranium-238 and their activation products. Each dosimeter block contains approximately 20 milligrams of neptunium-237 and 13 milligrams of uranium-238 contained in a 3/8-inch OD sealed brass tube. Each tube is placed in a

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

1/2-inch diameter hole in the dosimeter block (one neptunium-237 and one uranium-238 tube per block), and the space around the tube is filled with cadmium oxide. After placement of this material, each hole is blocked with two 1/16-inch aluminum spacer discs and an outer 1/8-inch steel cover disc, which is welded in place. Dosimeters of copper, nickel, aluminimum-cobalt and cadmium-shielded aluminum-cobalt are also secured in holes drilled in spacers located at the top, middle and bottom of each Type II capsule.

Capsule Type	Capsule Identification
I	S
II	V
II	T
I	U
II	X
I	W
I	Y
I	Z

3/4.4.10 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2, and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i).

Components of the Reactor Coolant System were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, January, 1970 Edition and the Winter Addenda of 1970.

The surveillance requirements for post-RCS opening, modifications and repairs ensure that RCS integrity is demonstrated following conditions that may have affected system integrity.

For normal opening the integrity of the system, in terms of strength, is unchanged. If the system does not leak at 2335 psig (operating pressure + 100 psi: ± 100 psi is normal system pressure fluctuation), it will be leak tight during normal operation.

For repairs on components greater than 2 in. diameter, the thorough nondestructive testing gives a very high degree of confidence in the integrity of the system, and will detect any significant defects in and near the new welds.

Repairs on components 2 in. in diameter or smaller are relatively minor in comparison and the surface examination assures a similar standard of integrity. In all cases, the leak test will ensure leak tightness during normal operation.

3/4.4.11 REACTOR COOLANT SYSTEM VENTS

Reactor Coolant System vents are provided to exhaust noncondensible gases and/or steam from the Reactor Coolant System that could inhibit natural circulation core cooling. The OPERABILITY of at least one Reactor Coolant System vent path from the reactor vessel head, and the pressurizer steam space ensures that the capability exists to perform this function.

REACTOR COOLANT SYSTEM VENTS (Continued)

The valve redundancy of the Reactor Coolant System vent paths serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve, power supply, or control system does not prevent isolation of the vent path. The performances of the specified surveillances will verify the operability of the system.

The function, capabilities, and testing requirements of the Reactor Coolant System vents are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

BASES

3/4.5.1 ACCUMULATORS

The OPERABILITY of each Reactor Coolant System (RCS) accumulator ensures that a sufficient volume of borated water will be immediately forced into the reactor core through each of the cold legs in the event the RCS pressure falls below the pressure of the accumulators. This initial surge of water into the core provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on accumulator volume, boron concentration and pressure ensure that the assumptions used for accumulator injection in the safety analysis are met.

Because the accumulator isolation valves fail to meet single failure criteria, removal of power to the valves is required.

The limits for operation with an accumulator inoperable for any reason except an isolation valve closed minimizes the time exposure of the plant to a LOCA event occurring concurrent with failure of an additional accumulator which may result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be immediately opened, the full capability of one accumulator is not available and prompt action is required to place the reactor in a mode where this capability is not required.

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of ECCS equipment and flow paths required in Modes 1, 2, 3 ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming any single failure consideration. Two SI pumps and one RHR pump operating in conjunction with two accumulators are capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all pipe break sizes up to and including the maximum hypothetical accident of a circumferential rupture of a reactor coolant loop. In addition, the RHR subsystem provides long-term core cooling capability in the recirculation mode during the accident recovery period.

With the RCS temperature below 350 F, operation with less than full redundant equipment is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

The Surveillance Requirements provided for each component ensures that ECCS OPERABILITY is maintained and verified periodically. Surveillance Requirements for testing of redundant equipment prior to maintenance on affected equipment ensure that at least one independent subsystem is OPERABLE prior to rendering the other inoperable.

3/4.5.4 REFUELING WATER STORAGE TANK

The OPERABILITY of the refueling water storage tank (RWST) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWST minimum volume and boron concentration ensure that: (1) sufficient water is available within containment following the injection phase to permit recirculation cooling flow to the core, and (2) the reactor will remain subcritical in the cold condition following mixing of the RWST and the RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The temperature limits on the RWST solution ensure that: 1) the solubility of the borated water will be maintained, and 2) the temperature of the RWST solution is consistent with the LOCA analysis.

3/4.6 CONTAINMENT SYSTEMS

BASES

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with the leakage rate limitation, will limit the SITE BOUNDARY radiation doses to within the dose guideline values of 10CFR Part 100 during accident conditions.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the safety analyses at the peak accident pressure, P_a, which would limit radiation at the site boundary to levels that are less than 10 CFR 100 limits. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a or 0.75 L_t, as applicable, during performance of the periodic test to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates is consistent with the requirements of Appendix J of 10 CFR Part 50.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provides assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that: (1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of 2.5 psi, and (2) the containment peak pressure does not exceed the design pressure of 59 psig during a design basis accident.

The maximum peak pressure expected to be obtained from a design basis accident event is 49.9 psig. The limit of 3 psig for initial positive containment pressure will limit the total pressure to 52.9 psig, which is less than design pressure and is consistent with the safety analyses.

3/4.6.1.5 AIR TEMPERATURE

The limitations on containment average air temperature ensure that the overall containment average air temperature does not exceed the initial temperature condition assumed in the safety analysis for a design basis accident. Measurement shall be made at all listed locations, whether by fixed or portable instruments, prior to determining the average air temperature.

3/4.6.1.6 CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the containment will withstand the maximum pressure of 52.9 psig in the event of a design basis accident (Ref. Bases 3/4.6.1.4). The measurement of containment tendon lift-off force, the tensile tests of the tendon wires or strands, the visual examination of tendons, anchorages and exposed interior and exterior surfaces of the containment, and the Type A leakage test are sufficient to demonstrate this capability.

The required Special Reports from any engineering evaluation of containment abnormalities shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, the results of the engineering evaluation, and the corrective actions taken.

3/4.6.1.7 CONTAINMENT VENTILATION SYSTEM

The containment purge supply and exhaust isolation valves are required to be closed during a Design Basis Accident. Maintaining these valves closed during plant operation ensures that excessive quantities of radioactive materials will not be released via the Containment Purge System. The total time the containment purge supply and exhaust isolation valves may be open during MODES 1, AND 2 in a calendar year is a function of anticipated need and operability experience.

Leakage integrity tests with a maximum allowable leakage rate for containment purge supply and exhaust supply valves will provide early indication of resilient material seal degradation and will allow opportunity for repair before gross leakage failures could develop. The 0.60 L_a leakage limit of Specification 3.6.1.2b. Shall not be exceeded when the leakage rates determined by the leakage integrity tests of these valves are added to the previously determined total for all other valves and penetrations subject to Type B and C tests.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 CONTAINMENT SPRAY SYSTEM

The OPERABILITY of the Containment Spray System ensures that containment depressurization and cooling capability will be available in the event of a design basis accident. The pressure reduction and resultant lower containment leakage rate are consistent with the assumptions used in the safety analyses.

The Containment Spray System and the Containment Cooling System are redundant to each other in providing post-accident cooling of the containment atmosphere. The allowable out-of-service time requirements for the Containment Spray System have been maintained consistent with that assigned other inoperable ESF equipment and do not reflect the additional redundancy in cooling capability provided by the Containment Cooling System.

3/4.6.2.2 CONTAINMENT COOLING SYSTEM

The OPERABILITY of the Containment Cooling System ensures that adequate heat removal capacity is available during post-LOCA conditions. The emergency containment coolers are a full capacity system and are redundant to the spray system in terms of the heat removal function for the design basis accident.

The Containment Cooling System and the Containment Spray System are redundant to each other in providing post-accident cooling of the containment atmosphere. The allowable out-of-service time requirements for the Containment Cooling System have been maintained consistent with that assigned other inoperable ESF equipment and do not reflect the additional redundancy in cooling capability provided by the Containment Spray System.

3/4.6.3 EMERGENCY CONTAINMENT FILTERING SYSTEM

The OPERABILITY of the Emergency Containment Filtering System ensures that sufficient iodine removal capability will be available in the event of a LOCA. The reduction in containment iodine inventory reduces the resulting SITE BOUNDARY radiation doses associated with containment leakage. The operation of this system and resultant iodine removal capacity are consistent with the assumptions used in the LOCA analyses. System components are not subject to rapid deterioration. Visual inspection and operating/performance tests after maintenance, prolonged operation, and at the required frequencies provide assurances of system reliability and will ensure conditions which could cause system failure. Filter performance tests are conducted in accordance with the methodology and intent of ANSI N510- 1975.

EMERGENCY CONTAINMENT FILTERING SYSTEM (Continued)

A Filter Efficiency of 90% is applicable to Turkey Point Safety Analysis. The HEPA filter efficiency combined with 99+% elemental iodine removal and 30% organic iodine efficiency for the activated carbon filter will give the applicable overall efficiency.

3/4.6.4 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment and is consistent with the requirements of General Design Criteria 54 through 57 of Appendix A to 10 CFR Part 50. Containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

3/4.6.5 COMBUSTIBLE GAS CONTROL MONITORS

The OPERABILITY of the Hydrogen Monitors ensures the detection of hydrogen buildup within containment following a LOCA to allow operator action to reduce the hydrogen concentration below its flammable limit.

3/4.6.6 POST ACCIDENT CONTAINMENT VENT SYSTEM

The OPERABILITY of the Post Accident Containment Vent System ensures the capability for emergency venting of containment following a LOCA to reduce the hydrogen concentration to below its flammable limit.

PACVS system components are not subject to rapid deterioration, having lifetimes of many years, even under continuous flow conditions. Visual inspection and operating tests provide assurance of system reliability and will ensure early detection of conditions which could cause the system to fail or operate improperly. The performance tests prove that filters have been properly installed, that no deterioration or damage has occurred, and that all components and subsystems operate properly. The tests are performed in accordance with the methodology and intent of ANSI N510-1975 and provide assurance that filter performance has not deteriorated below required specification values due to aging, contamination or other effects.

A filter efficiency of 99% is applicable to Turkey Point Safety Analysis. The HEPA filter efficiency combined with 90+ % methyliodide removal efficiency for the activated carbon filter will give the applicable overall efficiency.

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line Code safety valves ensures that the Secondary System pressure will be limited to within 110% (1193.5 psig) of its design pressure of 1085 psig during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a Turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code. The total relieving capacity for all valves on all of the steam lines is 10,670,400 lbs/hr which is 111% of the total secondary steam flow of 9,600,000 lbs/hr at 100% RATED THERMAL POWER.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in Secondary Coolant System steam flow and THERMAL POWER required by the reduced Reactor trip settings of the Power Range Neutron Flux channels. The Reactor Trip Setpoint reductions are derived on the following bases:

$$SP = \frac{(X) - (Y)(V)}{X} \times (109)$$

Where:

SP = Reduced Reactor Trip Setpoint in percent of RATED THERMAL POWER

V = Maximum number of inoperable safety valves per steam line

109 = Power Range Neutron Flux-High Trip Setpoint

X = Total relieving capacity of all safety valves per steam line in lbs/hour, and

Y = Maximum relieving capacity of any one safety valve in lbs/hour

3/4.7.1 TURBINE CYCLE

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM

The OPERABILITY of the Auxiliary Feedwater System ensures that the Reactor Coolant System can be cooled down to less than 350 F from normal operating conditions in the event of a total loss of off-site power. Steam can be supplied to the pump turbines from either or both units through redundant steam headers. Two D.C. motor operated valves and one A.C. motor operated valve on each unit isolate the three main steam lines from these headers. Both the D.C. and A.C. motor operated valves are powered from safety related sources. Auxiliary feedwater can be supplied through redundant lines to the safety related portions of the main feedwater lines to each of the steam generators. Air operated fail closed flow control valves are provided to modulate the flow to each steam generator. A qualified bottle nitrogen system automatically supplies motive power for the valves in the event of a failure in the instrument air system. Each steam driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 373 gpm to the entrance of the steam generators. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350 F when the Residual Heat Removal System may be placed into operation.

The monthly testing of the auxiliary feedwater pumps will verify their operability. Proper functioning of the turbine admission valve and the operation of the pumps will demonstrate the integrity of the system. Verification of correct operation will be made both from instrumentation within the control room and direct visual observation of the pumps.

3/4.7.1.3 CONDENSATE STORAGE TANKS

There are two(2) seismically designed 250,000 gallon condensate storage tanks. A minimum of 185,000 gallons is maintained in each tank. The OPERABILITY of the condensate storage tank with the minimum water volume ensures that sufficient water is available to maintain the Reactor Coolant System at HOT STANDBY conditions for approximately 23 hours or maintain the Reactor Coolant System at HOT STANDBY conditions for 15 hours and then cool down the Reactor Coolant System to below 350 F at which point the Residual Heat Removal System may be placed in operation.

3/4.7.1.4 SPECIFIC ACTIVITY

The limit on secondary coolant specific activity is based on a postulated release of secondary coolant equivalent to the contents of three steam generators to the atmosphere due to a net load rejection. The limiting dose for this case would result from radioactive iodine in the secondary coolant. One tenth of the iodine in the secondary coolant is assumed to reach the site boundary making allowance for plate-out and retention in water

SPECIFIC ACTIVITY (Continued)

droplets. The inhalation thyroid dose at the site boundary is then;

$$\text{Dose (Rem)} = C \circ V \circ B \circ DCF \circ X/Q \circ 0.1$$

Where:

C	=	secondary coolant dose equivalent I-131 specific activity
	=	0.2 curies/m^3 (μCi/cc) or 0.1 Ci/m^3, each unit
V	=	equivalent secondary coolant volume released = 214 m^3
B	=	breathing rate = 3.47×10^{-4} m^3/sec.
X/Q	=	atmospheric dispersion parameter = 1.54×10^{-4} sec/m^3
0.1	=	equivalent fraction of activity released
DCF	=	dose conversion factor, Rem/ci

The resultant thyroid dose is less than 1.5 Rem.

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVES

The OPERABILITY of the main steam line isolation valves ensures that no more than one steam generator will blow down in the event of a steam line rupture. This restriction is required to: (1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and (2) limit the pressure rise within containment in the event the steam line rupture occurs within containment. The OPERABILITY of the main steam isolation valves within the closure times of the Surveillance Requirements are consistent with the assumptions used in the safety analyses.

3/4.7.1.6 STANDBY FEEDWATER SYSTEM

The purpose of this specification and the supporting surveillance requirements is to provide for administrative controls which will assure availability and performance of the non-safety grade Standby Feedwater System. The term availability is used rather than operability so as to positively avoid any implication or connotation that the Standby Feedwater System is designed or documented to meet safety system requirements such as seismic loads, environmental qualification, safety grade emergency power supply, etc. The system does consist of commercial grade components designed and constructed to industry and FPL standards for this class of equipment located in the outdoor plant environment typical of FPL facilities system wide. The system is expected to perform with high reliability, i.e., comparable to that typically achieved with this class of equipment. FPL intends to maintain the system in good operating condition with regard to appearance, structures, supports, component maintenance, calibrations, etc.,

STANDBY FEEDWATER SYSTEM (Continued)

but would not document these activities to the degree required for safety grade systems.

The Standby Feedwater System can be used as a backup to the Auxiliary Feedwater (AFW) System in the event the AFW System does not function properly. The system would be manually started and controlled by the operator when needed. In the event of a loss of offsite power the pumps can be powered via the non-safety grade diesel generators connected to the non-vital 4160 volt bus.

A supply of 60,000 gallons from the Demineralized Water Storage Tank for the Standby Feedwater Pumps is sufficient water to remove decay heat from the reactor for six(6) hours for a single unit or two (2) hours for two units. This was the basis used for requiring 60,000 gallons of water in the non-safety grade Demineralized Water Storage Tank and is judged to provide sufficient time for restoring the AFW System or establishing make-up to the Demineralized Water Storage Tank.

The motor driven Standby Feedwater Pumps are not designed to NRC requirements applicable to Emergency Feedwater Systems and not required to meet design basis events. These pumps may be out of service for up to 24 hours before initiating formal notification because of the extremely low probability of such an event.

The guidelines for NRC notification in case of both pumps being out of service for longer than 24 hours are provided in applicable plant procedures, as a VOLUNTARY 4 hour notification.

Adequate demineralized water for the standby feedwater system will be verified once per 24 hours. The Demineralized Water Storage Tank provides a source of water to several systems and therefore, requires daily verification.

The standby feedwater pumps will be verified available monthly on a STAGGERED TEST BASIS by starting and operating them in the recirculation mode typically from their normal power supply. Also, during each unit's refueling outage, the respective standby feedwater pump will be powered from the unit's C bus utilizing Units 1 and 2 non-safety grade diesel generators and flow tested to the nuclear unit's steam generators. Prior to this test, the refueling unit's C bus will be de-energized and the necessary loads will be transferred to the other unit's C bus.

This surveillance regimen will thus demonstrate availability of the entire flow path, backup non-safety grade power supply and pump associated with a unit at least each refueling outage. The pump, motor driver, and normal power supply availability would typically be demonstrated by operation of the pumps in the recirculation mode monthly on a staggered test basis.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on steam generator pressure and temperature ensures that the pressure-induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations of 70 F and 200 psig are based on a steam generator RT_{NDT} of 60 F and are sufficient to prevent brittle fracture.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

The OPERABILITY of the Component Cooling Water System ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The redundant cooling capacity provided by the required equipment ensures that cooling capacity is consistent with assumptions in the safety analysis, assuming a single failure. Analysis results have shown that one pump and 2 heat exchangers will meet cooling requirements assumed in the accident analysis.

3/4.7.4 INTAKE COOLING WATER SYSTEM

The OPERABILITY of the Intake Cooling Water System ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the safety analyses.

3/4.7.5 ULTIMATE HEAT SINK

The limitation of the ultimate heat sink (Intake Cooling Water) temperature ensures that initial conditions assumed in the safety analysis for cooling capacity are not exceeded.

The design basis require that the ultimate heat sink system must be capable of removing the design basis heat load while keeping the Component Cooling Water (CCW) heat exchanger return temperature at or below 125 F. The capability of removing heat from CCW is dependent on the heat transfer coefficient of the heat exchangers. For a higher heat transfer coefficient and/or higher intake supply flow rate, a temperature greater than 95 F can be justified. Periodic testing of the CCW heat exchangers to determine the heat transfer coefficient is performed to plot performance characteristics of the heat exchangers.

Portable temperature monitors may be used to measure the intake cooling water temperature.

3/4.7.6 FLOOD PROTECTION

The limitation on flood protection ensures that facility protective actions will be taken in the event of flood conditions. The installation of the stoplogs ensures adequate protection for wave run-up effects where no permanent adjacent structures exist and provides protection to safety-related equipment. The maximum wave run-up from the probable maximum hurricane (PMH) has been calculated to be elevation 22.5 feet Mean Low Water (MLW) to the east, and 20 feet above MLW on the remaining sides.

3/4.7.7 CONTROL ROOM EMERGENCY AIR CLEANUP

The OPERABILITY of the Control Room Ventilation System ensures that: (1) the ambient air temperature does not exceed the allowable temperature for continuous-duty rating for the equipment and instrumentation cooled by this system, and (2) the control room will remain habitable for operations personnel during and following all credible accident conditions. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criterion 19 of Appendix A, 10 CFR Part 50.

System components are not subject to rapid deterioration, having lifetimes of many years, even under continous flow conditions. Visual inspection and operating tests provide assurance of system reliability and will ensure early detection of conditions which could cause the system to fail or operate improperly. The filters performance tests prove that filters have been properly installed, that no deterioration or damage has occurred, and that all components and subsystems operate properly. The tests are performed in accordance with the methodology and intent of ANSI N510 (1975) and provide assurance that filter performance has not deteriorated below returned specification values due to aging, contamination, or other effects.

3/4.7.8 SNUBBERS

All snubbers are required OPERABLE to ensure that the structural integrity of the reactor coolant system and all other Safety-Related Systems is maintained during and following a seismic or other event initiating dynamic loads. Snubbers excluded from this inspection program are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed, would have no adverse effect on any Safety-Related Systems.

SNUBBERS (Continued)

The visual inspection frequency is based upon maintaining a constant level of snubber protection to systems. Therefore, the required inspection interval varies inversely with the observed snubber failures and is determined by the number of inoperable snubbers found during an inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shortened inspection interval will override the previous schedule.

When the cause of the rejection of a snubber by visual inspection is clearly established and remedied for the snubber and for any other snubbers that may be generically susceptible, and verified operable by inservice functional testing, that snubber may be exempted from being counted as inoperable for the purposes of establishing the next visual inspection interval. Generically susceptible snubbers are those which are of a specific make or model and have the same design features directly related to rejection of the snubber by visual inspection, or are similarly located or exposed to the same environmental conditions such as temperature, radiation, and vibration.

When a snubber is found inoperable, an evaluation is performed, in addition to the determination of the snubber mode of failure, in order to determine if any Safety Related System or component has been adversely affected by the inoperability of the snubber. The evaluation shall determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the supported component or system.

To provide assurance of snubber functional reliability, a representative sample of the installed snubbers will be functionally tested during plant refueling SHUTDOWNS. Observed failure of these sample snubbers shall require functional testing of additional units.

In cases where the cause of the functional failure has been identified additional testing shall be based on manufacturer's or engineering recommendations. As applicable, this additional testing increases the probability of locating possible inoperable snubbers without testing 100% of the safety-related snubbers.

The service life of a snubber is evaluated via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubber, spring replaced, in high radiation area, in high temperature area, etc.). The

SNUBBERS (Continued)

requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life. The requirements for the maintenance of records and the snubber service life review are not intended to affect plant operation.

3/4.7.9 SEALED SOURCE CONTAMINATION

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(a)(3) limits for plutonium. This limitation will ensure that leakage from Byproduct, Source, and Special Nuclear Material sources will not exceed allowable intake values.

Sealed sources are classified into three groups according to their use, with Surveillance Requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism (i.e., sealed sources within radiation monitoring or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shielded mechanism.

3/4.8 ELECTRICAL POWER SYSTEMS BASES

3/4.8.1, 3/4.8.2 AND 3/4.8.3 A.C. SOURCES , D.C. SOURCES, AND ONSITE POWER DISTRIBUTION

LATER

3/4.9 REFUELING OPERATIONS

BASES

3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: (1) the reactor will remain subcritical during CORE ALTERATIONS, and (2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the safety analyses. The locking closed of the required valves during refueling operations precludes the possibility of uncontrolled boron dilution of the filled portion of the RCS. This action prevents flow to the RCS of unborated water by closing flow paths from sources of unborated water.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the Source Range Neutron Flux Monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short-lived fission products.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

The requirements on containment building penetration closure and OPERABILITY ensure that a release of radioactive material within containment will be restricted from leakage to the environment. The OPERABILITY and closure restrictions are sufficient to restrict radioactive material release from a fuel element rupture based upon the lack of containment pressurization potential while in the REFUELING MODE.

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity conditions during CORE ALTERATIONS.

3/4.9.6 MANIPULATOR CRANE

The OPERABILITY requirements for the manipulator crane ensure that: (1) manipulator crane will be used for movement of drive rods and fuel assemblies, (2) each crane has sufficient load capacity to lift a drive rod or fuel assembly, and (3) the core internals and reactor vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE AREAS

The restriction on movement of loads in excess of the nominal weight of a fuel and control rod assembly and associated handling tool over other fuel assemblies in the storage pool ensures that in the event this load is dropped: (1) the activity release will be limited to that contained in a single fuel assembly, and (2) any possible distortion of fuel in the storage racks will not result in a critical array. This assumption is consistent with the activity release assumed in the safety analyses.

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

The requirement that at least one residual heat removal (RHR) loop be in operation ensures that: (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor vessel below 140 F as required during the REFUELING MODE, and (2) sufficient coolant circulation is maintained through the core to minimize the effect of a boron dilution incident and prevent boron stratification.

The requirement to have two RHR loops OPERABLE when there is less than 23 feet of water above the reactor vessel flange ensures that a single failure of the operating RHR loop will not result in a complete loss of residual heat removal capability. With the reactor vessel head removed and at least 23 feet of water above the reactor pressure vessel flange, a large heat sink is available for core cooling. Thus, in the event of a failure of the operating RHR loop, adequate time is provided to initiate emergency procedures to cool the core.

3/4.9.9 CONTAINMENT VENTILATION ISOLATION SYSTEM

The OPERABILITY of this system ensures that the containment ventilation penetrations will be automatically isolated upon detection of high radiation levels within the containment. The OPERABILITY of this system is required to restrict the release of radioactive material from the containment atmosphere to the environment.

3/4.9.10 and 3/4.9.11 WATER LEVEL - REACTOR VESSEL AND STORAGE POOL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the safety analysis.

3/4.9.12 HANDLING OF SPENT FUEL CASK

Limiting spent fuel decay time to a minimum of 1,525 hours prior to moving a spent fuel cask into the spent fuel pit will ensure that potential offsite doses are a fraction of 10 CFR Part 100 limits should a dropped cask strike the stored fuel assemblies.

The restriction to allow only a single element cask to be moved into the spent fuel pit will ensure the maintenance of water inventory in the unlikely event of an uncontrolled cask descent. Use of a single element cask which nominally weighs about twenty-five tons will also increase crane safety margins by about a factor of four.

Requiring that spent fuel decay time be at least 120 days prior to moving a fuel assembly outside the fuel storage pit in a shipping cask will ensure that potential offsite doses are a fraction of 10 CFR 100 limits should a dropped cask and ruptured fuel assembly release activity directly to the atmosphere.

3/4.9.13 RADIATION MONITORING

The OPERABILITY of the containment radiation monitors ensures continuous monitoring of radiation levels to provide immediate indication of an unsafe condition.

3/4.9.14 SPENT FUEL STORAGE

1. The spent fuel storage racks provide safe subcritical storage of fuel assemblies by providing sufficient center-to-center spacing or a combination of spacing and poison to assure k_{eff} is equal to or less than 0.95 for normal operations and postulated accidents.

SPENT FUEL STORAGE (Continued)

2.* The spent fuel racks are divided into two regions. Region I racks have a 10.6 inch center-to-center spacing and the Region II racks have a 9.0 inch center-to-center spacing. Because of the larger center-to-center spacing and poison (B^{10}) concentration of Region I cells, the only restriction for placement of fuel is that the initial fuel assembly enrichment is equal to or less than 4.5 weight percent of U-235. The limiting value of U-235 enrichment is based upon the assumptions in the spent fuel safety analyses and assures that the limiting criteria for criticality is not exceeded. Prior to placement in Region II cell locations, strict controls are employed to evaluate burnup of the spent fuel assembly. Upon determination that the fuel assembly meets the burnup requirements of Table 3.9-1, placement in a Region II cell is authorized. These positive controls assure the fuel enrichment limits assumed in the safety analyses will not be exceeded.

* This Technical Specification is applicable upon installation of the new two-region high density spent fuel racks.

3/4.10.1 SHUTDOWN MARGIN

This special test exception provides that a minimum amount of control rod worth is immediately available for reactivity control when tests are performed for control rod worth measurement. This special test exception is required to permit the periodic verification of the actual versus predicted core reactivity condition occurring as a result of fuel burnup or fuel cycling operations.

3/4.10.2 GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

This special test exception permits individual control rods to be positioned outside of their normal group heights and insertion limits during the performance of such PHYSICS TESTS as those required to measure control rod worth.

3/4.10.3 PHYSICS TESTS

This special test exception permits PHYSICS TESTS to be performed at less than or equal to 5% of RATED THERMAL POWER with the RCS Tavg slightly lower than normally allowed so that the fundamental nuclear characteristics of the core and related instrumentation can be verified. In order for various characteristics to be accurately measured, it is at times necessary to operate outside the normal restrictions of these Technical Specifications. For instance, to measure the moderator temperature coefficient at BOL, it is necessary to position the various control rods at heights which may not normally be allowed by Specification 3.1.3.6.

3/4.10.4 REACTOR COOLANT LOOPS

This special test exception permits reactor criticality under no flow conditions and is required to perform certain PHYSICS TESTS while at low THERMAL POWER levels.

3/4.10.5 POSITION INDICATION SYSTEM - SHUTDOWN

This special test exception permits the Position Indication Systems to be inoperable during rod drop time measurements. The exception is required since the data necessary to determine the rod drop time are derived from the induced voltage in the position indicator coils as the rod is dropped. This induced voltage is small compared to the normal voltage and, therefore, cannot be observed if the Position Indication Systems remain OPERABLE.

3/4.11 RADIOACTIVE EFFLUENTS

BASES

3/4.11.1 LIQUID EFFLUENTS

3/4.11.1.1 CONCENTRATION

This specification is provided to ensure that the concentration of radioactive materials released in liquid waste effluents to UNRESTRICTED AREAS will be less than the concentration levels specified in 10 CFR Part 20, Appendix B, Table II, Column 2. This limitation provides additional assurance that the levels of radioactive materials in bodies of water in UNRESTRICTED AREAS will result in exposures within: (1) the Section II.A design objective of Appendix I, 10 CFR Part 50, to a MEMBER OF THE PUBLIC, and (2) the limits of 10 CFR Part 20.106(e) to the population. The concentration limit for dissolved or entrained noble gases is based upon the assumption that Xe-135 is the controlling radioisotope and its MPC in air (submersion) was converted to an equivalent concentration in water using the methods described in International Commission on Radiological Protection (ICRP) Publication 2.

The specification applies to the release of radioactive materials in liquid effluents from all units at the site.

The required detection capabilities for radioactive materials in liquid waste samples are tabulated in terms of the lower limits of detection (LLDs). Detailed discussion of the LLD, and other detection limits can be found in HASL procedures Manual, HASL-300 (revised annually), Currie, L.A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry," Anal. Chem. 40, 586-93 (1968), and Hartwell, J.K., "Detection Limits for Radioanalytical Counting Techniques, " Atlantic Richfield Hanford Company Report ARH-SA-215 (June 1975).

3/4.11.1.2. DOSE

This specification is provided to implement the requirements of Sections II.A, III.A, and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operating implements the guides set forth in Section II.A of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in liquid effluents to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable." The dose calculation methodology and parameters in the ODCM implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by

DOSE (Continued)

calculational procedures based on models and data, such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The equations specified in the ODCM for calculating the doses due to the actual release rates of radioactive materials in liquid effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I", March, 1976, and Regulatory Guide 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I", April, 1977.

This specification applies to the release of radioactive materials in liquid effluents from each unit at the site. When shared Radwaste Treatment Systems are used by more than one unit on a site, the wastes from all units are mixed for shared treatment: by such mixing, the effluent releases cannot accurately be ascribed to a specific unit. An estimate should be made of the contributions from each unit based on input conditions, e.g., flow rates and radioactivity concentrations, or, if not practicable, the treated effluent releases may be allocated equally to each of the radioactive waste producing units sharing the Radwaste Treatment System. For determining conformance to LCOs, these allocations from shared Radwaste Treatment Systems are to be added to the releases specifically attributed to each unit to obtain the total releases per unit.

3/4.11.1.3 LIQUID RADWASTE TREATMENT SYSTEM

The OPERABILITY of the Liquid Radwaste Treatment System ensures that this system will be available for use whenever liquid effluents require treatment prior to release to the environment. The requirement that the appropriate portions of this system be used when specified provides assurance that the releases of radioactive materials in liquid effluents will be kept "as low as is reasonably achievable." This specification implements the requirements of 10 CFR 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50 and the design objective given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the Liquid Radwaste Treatment System were specified as a suitable fraction of the dose design objectives set forth in Section II.A of Appendix I, 10 CFR Part 50, for liquid effluents.

LIQUID RADWASTE TREATMENT SYSTEM (Continued)

This specification applies to the release of radioactive materials in liquid effluents from each unit at the site. When shared Radwaste Treatment Systems are used by more than one unit on a site, the wastes from all units are mixed for shared treatment; by such mixing, the effluent releases cannot accurately be ascribed to a specific unit. An estimate should be made of the contributions from each unit based on input conditions, e.g., flow rates and radioactivity concentrations, or, if not practicable, the treated effluent releases may be allocated equally to each of the radioactive waste producing units sharing the Radwaste Treatement System. For determining conformance to LCOs, these allocations from shared Radwaste Treatment Systems are to be added to the releases specifically attributed to each unit to obtain the total releases per unit.

3/4.11.2 GASEOUS EFFLUENTS

3/4.11.2.1 DOSE RATE

This specification is provided to ensure that the dose at any time at and beyond the SITE BOUNDARY from gaseous effluents from all units on the site will be within the annual dose limits of 10 CFR Part 20 to UNRESTRICTED AREAS. The annual dose limits are the doses associated with the concentrations of 10 CFR Part 20, Appendix B, Table II, Column 1. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of a MEMBER OF THE PUBLIC in an UNRESTRICTED AREA, either within or outside the SITE BOUNDARY, to annual average concentrations exceeding the limits specified in Appendix B, Table II of 10CFR Part 20 (10 CFR Part 20.106(b)). For MEMBERS OF THE PUBLIC who may at times be within the SITE BOUNDARY, the occupancy of that MEMBER OF THE PUBLIC will usually be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the SITE BOUNDARY. Examples of calculations for such MEMBERS OF THE PUBLIC, with the appropriate occupancy factors, shall be given in the ODCM. The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above background to a MEMBER OF THE PUBLIC at or beyond the SITE BOUNDARY to less than or equal to 500 mrems/year to the whole body or to less than or equal to 3000 mrems/year to the skin. These release rate limits also restrict, at all times, the corresponding thyroid dose rate above background to a child via the inhalation pathway to less than or equal to 1500 mrems/year.

DOSE RATE (Continued)

The required detection capabilities for radioactive materials in gaseous waste samples are tabulated in terms of the lower limits of detection (LLDs). Detailed discussion of the LLD, and other detection limits can be found in HASL Procedures Manual, HASL 300 (revised annually), Currie, L.A., "Limits for Qualitative Detection and Quantitative Determination-Application to Radiochemistry," Anal. Chem. 40, 586-93 (1968), and Hartwell, J.K., "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARE-SA-215 (June 1975).

3/4.11.2.2 DOSE - NOBLE GASES

This specification is provided to implement the requirements of Sections II.B, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.B of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in gaseous effluents to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable." The Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The dose calculation methodology and parameters established in the ODCM for calculating the doses due to the actual release rates of radioactive noble gases in gaseous effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I, "Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-water Cooled Reactors," Revision 1, July 1977. The ODCM equations provided for determining the air doses at and beyond the SITE BOUNDARY are based upon the historical average atmospheric conditions.

This specification applies to the release of radioactive materials in gaseous effluents from each unit at the site. When shared Radwaste Treatment Systems are used by more than one unit on a site, the wastes from all units are mixed for shared treatment; by such mixing, the effluent releases cannot accurately be ascribed to a specific unit. An estimate should be made of the contributions from each unit based on input conditions, e.g., flow rates and radioactivity concentrations, or, if not practicable, the treated effluent releases may be allocated equally to each of the radioactive waste producing units sharing the Radwaste Treatment System. For determining conformance to LCOs, these allocations from shared Radwaste Treatment Systems are to be added to the releases specifically attributed to each unit to obtain the total releases per unit.

3/4.11.2.3 DOSE - IODINE-131, IODINE-133, TRITIUM, AND RADIOACTIVE MATERIAL IN PARTICULATE FORM

This specification is provided to implement the requirements of Sections II.C, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Conditions for Operation are the guides set forth in Section II.C of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive materials in gaseous effluents to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable." The ODCM calculational methods specified in the Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data, such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The ODCM calculational methodology and parameters for calculating the doses due to the actual release rates of the subject materials are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Revision 1, July 1977. These equations also provide for determining the actual doses based upon the historical average atmospheric conditions. The release rate specifications for Iodine-131, Iodine-133, tritium and radionuclides in particulate form with half-lives greater than 8 days are dependent upon the existing radionuclide pathways to man in the areas at and beyond the SITE BOUNDARY. The pathways that were examined in the development of the calculations were: (1) individual inhalation of airborne radionuclides, (2) deposition of radionuclides onto green leafy vegetation with subsequent consumption by man, (3) deposition onto grassy areas where milk animals and meat producing animals graze with consumption of the milk and meat by man, and (4) deposition on the ground with subsequent exposure of man.

This specification applies to the release of radioactive materials in gaseous effluents from each unit at the site. When shared Radwaste Treatment Systems are used by more than one unit on a site, the wastes from all units are mixed for shared treatment; by such mixing, the effluent releases cannot accurately be ascribed to a specific unit. An estimate should be made of the contributions from each unit based on input conditions, e.g., flow rates and radioactivity concentrations, or, if not practicable, the treated effluent releases may be allocated equally to each of the radioactive waste producing units sharing the Radwaste Treatment System. For determining conformance to LCOs, these allocations from shared Radwaste Treatment Systems are to be added to the releases specifically attributed to each unit to obtain the total releases per unit.

3/4.11.2.4 GASEOUS RADWASTE TREATMENT SYSTEM

The OPERABILITY of the GAS DECAY TANK SYSTEM and the VENTILATION EXHAUST TREATMENT SYSTEM ensures that the systems will be available for use whenever gaseous effluents require treatment prior to release to the environment. The requirement that the appropriate portions of these systems be used, when specified, provides reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable." This specification implements the requirements of 10 CFR 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50 and the design objectives given in Section II.D of Appendix I to 10 CFR Part 50. The limits governing the use of appropriate portions of the systems were specified as a suitable fraction of the dose design objectives set forth in Sections II.B and II.C of Appendix I, 10 CFR Part 50, for gaseous effluents.

This specification applies to the release of radioactive materials in gaseous effluents from each unit at the site. When shared Radwaste Treatment Systems are used by more than one unit on a site, the wastes from all units are mixed for shared treatment; by such mixing, the effluent releases cannot accurately be ascribed to a specific unit. An estimate should be made of the contributions from each unit based on input conditions, e.g., flow rates and radioactivity concentrations, or, if not practicable, the treated effluent releases may be allocated equally to each of the radioactive waste producing units sharing the Radwaste Treatment System. For determining conformance to LCOS, these allocations from shared Radwaste Treatment Systems are to be added to the releases specifically attributed to each unit to obtain the total releases per unit.

3/4.11.2.5 EXPLOSIVE GAS MIXTURE

This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the GAS DECAY TANK SYSTEM is maintained below the flammability limits of hydrogen and oxygen. Maintaining the concentration of hydrogen and oxygen below their flammability limits provides assurance that the releases of radioactive materials will be controlled in conformance with the requirement of General Design Criterion 60 of Appendix A to 10 CFR Part 50.

3/4.11.2.6 GAS DECAY TANKS

The tanks included in this specification are those tanks for which the quantity of radioactivity contained is not limited directly or indirectly by another Technical Specification. Restricting the quantity of radioactivity contained in each gas decay tank provides assurance that in the event of an uncontrolled release of the tank's contents, the resulting whole body exposure to a MEMBER OF THE PUBLIC at the nearest SITE BOUNDARY will not exceed 0.5 rem.

3/4.11.3 SOLID RADIOACTIVE WASTES

This specification implements the requirements of 10 CFR 50.36a and General Design Criterion 60 of Appendix A to 10 CFR Part 50. The process parameters included in establishing the PROCESS CONTROL PROGRAM may include, but are not limited to, waste type, waste pH, waste/liquid/SOLIDIFICATION agent/catalyst ratios, waste oil content, waste principal chemical constituents, and mixing and curing times.

3/4.11.4 TOTAL DOSE

This specification is provided to meet the dose limitations of 40 CFR Part 190 that have been incorporated into 10 CFR Part 20 by 46 FR 18525. The specification requires the preparation and submittal of a Special Report whenever the calculated doses due to releases of radioactivity and to radiation from uranium fuel cycle sources exceed 25 mrems to the whole body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrems. For sites containing up to four reactors, it is highly unlikely that the resultant dose to a MEMBER OF THE PUBLIC will exceed the dose limits of 40 CFR Part 190 if the individual reactors remain within twice the dose design objectives of Appendix I, and if direct radiation doses from the units are kept small. The Special Report will describe a course of action that should result in the limitation of the annual dose to a MEMBER OF THE PUBLIC to within the 40 CFR Part 190 limits. For the purposes of the Special Report, it may be assumed that the dose commitment to the MEMBER OF THE PUBLIC from other uranium fuel cycle sources is negligible, with the exception that dose contributions from other nuclear fuel cycle facilities at the same site or within a radius of 8 km must be considered. If the dose to any MEMBER OF THE PUBLIC is estimated to exceed the requirements of 40 CFR Part 190, the Special Report with a request for a variance (provided the release conditions resulting in violation of 40 CFR Part 190 has not already been corrected), in accordance with the provisions of 40 CFR 190.11 and 10 CFR 20.405c, is considered to be a timely request and fulfills the requirements of 40 CFR Part 190 until NRC staff action is completed. The variance only relates to the limits of 40 CFR Part 190, and does not apply in any way to the other requirements for dose limitation of 10 CFR Part 20, as addressed in Specifications 3.11.1.1 and 3.11.2.1. An individual is not considered a MEMBER OF THE PUBLIC during any period in which he/she is engaged in carrying out any operation that is part of the nuclear fuel cycle.

3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

BASES

3/4.12.1 MONITORING PROGRAM

The Radiological Environmental Monitoring Program required by this specification provides representative measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides that lead to the highest potential radiation exposures of MEMBERS OF THE PUBLIC resulting from the plant operation. This monitoring program implements Section IV.B.2 of Appendix I to 10 CFR part 50 and thereby supplements the Radiological Effluent Monitoring Program by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and the modeling of the environmental exposure pahtways.

The required detection capabilities for environmental sample analyses are tabulated in terms of the lower limits of detection (LLDs). The LLDs required by Table 4.12-1 are considered optimum for routine environmental measurements in industrial laboratories. It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement.

Detailed discussion of the LLD, and other detection limits, can be found in HASL Procedures Manual, HASL-300 (revised annually), Currie, L.A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry," Anal. Chem. 40, 586-93 (1968), and Hartwell, J. K., "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-215 (June 1975).

3/4.12.2 LAND USE CENSUS

This specification is provided to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the Radiological Environmental Monitoring Program given in the ODCM are made if required by the results of this census. The best information from the door-to-door survey, from aerial survey or from consulting with local agricultural authorities shall be used. This census satisfies the requirements of Section IV.B.3 of Appendix I to 10 CFR Part 50. Restricting the census to gardens of greater than 50 m^2 provides assurance that significant exposure pathways via leafy vegetables will be identified and monitored since a garden of this size is the minimum required to produce the quantity (26 kg/year) of leafy vegetables assumed in Regulatory Guide 1.109 for consumption by a child. To determine this minimum garden size, the following assumptions were made: (1) 20% of the garden was used for growing broad leaf vegetation (i.e., similar to lettuce and cabbage), and (2) a vegetation yield of 2 kg/m^2.

3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

The requirement for participation in an approved Interlaboratory Comparison Program is provided to ensure that independent checks on the precision and accuracy of the measurements of radioactive material in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring in order to demonstrate that the results are valid for the purposes of Section IV.B.2 of Appendix I to 10 CFR Part 50. This condition is satisfied by participation in the Environmental Radioactivity Laboratory Intercomparison Studies Program conducted by the Environmental Protection Agency (EPA).

SECTION 5.0

DESIGN FEATURES

5.0 DESIGN FEATURES

5.1 SITE

EXCLUSION AREA

5.1.1 The Exclusion Area is as shown in Figure 5.1-1.

LOW POPULATION ZONE

5.1.2 The Low Population Zone is as shown in Figure 5.1-1.

MAP DEFINING UNRESTRICTED AREAS AND SITE BOUNDARY FOR RADIOACTIVE GASEOUS AND LIQUID EFFLUENTS

5.1.3 Information regarding radioactive gaseous and liquid effluents, which allows identification of structures and release points, SITE BOUNDARY, as well as definition of UNRESTRICTED AREAS within the Exclusion Area that are accessible to MEMBERS OF THE PUBLIC, is shown in Figure 5.1-1.

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The containment building is a steel-lined, reinforced concrete building of cylindrical shape, with a dome roof and having the following design features:

a. Nominal inside diameter = 116 feet.

b. Nominal inside height = 170.6 feet.

c. Minimum thickness of concrete walls = 3.75 feet.

d. Minimum thickness of concrete roof = 3.25 feet.

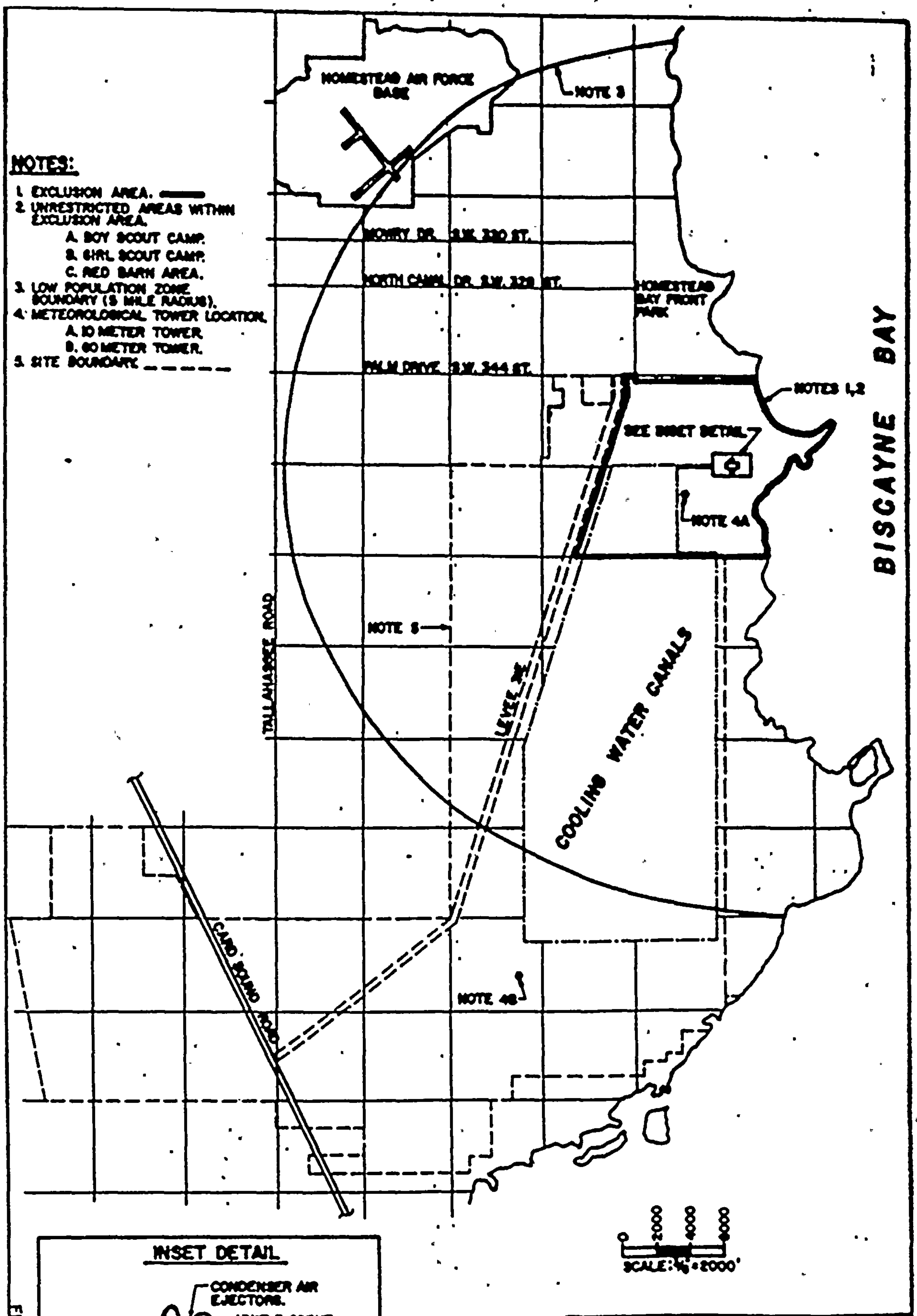
e. Minimum thickness of concrete floor pad = 10.5 feet.

f. Nominal thickness of steel liner = 0.25 inches.

g. Net free volume = 1,550,000 cubic feet.

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The containment building is designed and shall be maintained for a maximum internal pressure of 59 psig and a temperature of 283°F. The containment building is also structurally designed to withstand an internal vacuum of 2.5 psig.



FLORIDA POWER & LIGHT CO.
TURKEY POINT PLANT

SITE AREA MAP

FIGURE 5.1-1

5.0 DESIGN FEATURES

5.2 CONTAINMENT

STRUCTURAL LOADS

5.2.3 The containment is designed to withstand the loads resulting from an earthquake producing 0.05g in the horizontal and 0.033g in the vertical planes while at design pressure.

PENETRATIONS

5.2.4.1 Access doors and hatches are double gasketed. Electrical penetration canisters have double barriers. Piping penetrations are of the rigid welded type anchored to the containment wall.

5.2.4.2 The Containment Isolation valve actuation system is designed such that no single component failure will prevent containment isolation if required. Containment isolation is initiated by an automatic safety injection signal. In addition, the Containment Ventilation system valves close in response to the Containment Radiation Alarm.

CONTAINMENT SYSTEMS

5.2.5.1 The containment has two air circulation and cooling systems. Two CRDM coolers and four containment coolers (fan cooler units with centrifugal fans and water cooled heat exchangers) are installed for ambient temperature control during normal operation. Three emergency containment coolers (axial fans and water cooled heat exchangers) are installed for cooling the air-steam mixture resulting from a MHA.

5.2.5.2 The containment has an internal spray system which is capable of providing a distributed boric acid spray.

5.2.5.3 The containment has three emergency filtering units for removal of particulates and iodine resulting from a MHA.

CONTAINMENT FUNCTIONS

5.2.6 The containment completely encloses the Reactor Coolant System and ensures that an acceptable upper limit for leakage of radioactive materials to the environment is not exceeded even if gross failure of the Reactor Coolant System occurs. The structure provides biological shielding for both normal and accident situations.

5.0 DESIGN FEATURES

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The core shall contain 157 fuel assemblies with each fuel assembly initially containing 204 fuel rods clad with Zircaloy-4. The reactor core contains approximately 71 metric tons of uranium in the form of natural or slightly enriched uranium dioxide pellets. Each fuel rod shall have a nominal active fuel length of 144 inches.

CONTROL ROD ASSEMBLIES

5.3.2 The core shall contain 45 full-length control rod assemblies. The full-length control rod assemblies shall contain a nominal 142 inches of absorber material. The absorber material shall be silver, indium, and cadmium. All control rods shall be clad with stainless steel tubing.

BURNABLE POISON ROD ASSEMBLIES

5.3.3 Burnable poisons are in the form of rod clusters which are located in vacant rod cluster control guide tubes, or integral to the fuel design and are used for reactivity and/or power distribution control.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The Reactor Coolant System is designed and shall be maintained:

a. In accordance with the Code requirements specified in Section 4.1 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650 F except for the pressurizer which is 680 F.

VOLUME

5.4.2 The nominal total water and steam volume of the Reactor Coolant System is 9343 cubic feet at T_{avg} 574 F.

5.0 DESIGN FEATURES

5.4 REACTOR COOLANT SYSTEM

SEISMIC STRESS

5.4.3 All piping, components and supporting structures of the Reactor Coolant System are designed to Class I requirements. The design will withstand a seismic ground acceleration, 0.05g acting in the horizontal and 0.033g acting in the vertical planes simultaneously, with stresses maintained within code allowable working stresses and the maximum potential seismic ground acceleration 0.15g acting in the horizontal and 0.10g acting in the vertical planes simultaneously with no function loss.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological towers are located as shown in Figure 5.1-1.

5.6 FUEL STORAGE

5.6.1 CRITICALITY

5.6.1.1 The spent fuel storage racks are designed to provide safe subcritical storage of fuel assemblies by providing sufficient center-to-center spacing or a combination of spacing and poison and shall be maintained with:

a. A k_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, which includes a conservative allowance of 2.55% $\Delta k/k$ for uncertainties for single region spent fuel storage racks.

b. A k_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, which includes a conservative allowance in region 1 of 2.66% Δ k/k and in region 2 of 2.94% Δ k/k for uncertainties for two region fuel storage racks.

c. A nominal 13.7 inch center-to-center distance between fuel assemblies placed in the single-region storage racks. A nominal 10.6 inch center-to-center distance for Region 1 and 9.0 inch center-to-center distance for Region 2 for two region fuel storage racks.

5.6.1.2 The k_{eff} for new fuel stored in the new fuel storage racks shall not exceed 0.98 when the optimum moderation condition is assumed and equal to or less than 0.95 for fully flooded conditions.

5.6.1.3 Fuel assemblies placed in the spent fuel storage racks shall contain no more than 4.1 weight percent of U-235 when stored in a single-region spent fuel storage rack. After installation of the two-region high density spent fuel racks, the maximum enrichment loading for new fuel assemblies in the Region 1 spent fuel storage areas is 4.5 weight percent of U-235. Fuel assemblies placed in the new fuel storage racks shall contain no more than 4.5 weight percent of U-235.

5.6.1.4 Credit for burnup is taken in determining placement locations for spent fuel in the two-region spent fuel racks.* Administrative controls are employed to evaluate the burnup of each spent fuel assembly stored in areas where credit for burnup is taken. The burnup of spent fuel is ascertained by careful analysis of burnup history, prior to placement into the storage locations. Procedures shall require an independent check of the analysis of suitability for storage. A complete record of such analysis is kept for the time period that the spent fuel assembly remains in storage onsite.

DRAINAGE

5.6.2 The spent fuel storage pit is designed and shall be maintained to prevent inadvertent draining of the pool below a level of 6 feet above the fuel assemblies in the storage racks.

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 621** fuel assemblies in one region storage racks or 1404 in two region storage racks

STRUCTURE

5.6.4 The new and spent fuel pit structures are designed to withstand the anticipated earthquake loadings as Class 1 structures. Each spent fuel pit has a stainless steel liner to ensure against leakage.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7-1.

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* During rack installation, it will be necessary to temporarily store Region I fuel in the Region II spent fuel racks. Administrative controls will be utilized to maintain a checkerboard storage configuration, i.e., alternate cell occupation, in the Region II racks.

** The fuel assembly storage capacity for Unit 4 single region storage racks is 614.

TABLE 5.7-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System	200 heatup cycles at ≤ 100 F/h and 200 cooldown cycles at ≤ 100 F/h	Heatup cycle - T_{avg} from ≤ 200 F to ≥ 550 F. Cooldown cycle - T_{avg} from ≥ 550 F to ≤ 200 F.
	200 pressurizer cooldown cycles at ≤ 200 F/h	Pressurizer cooldown cycle temperatures from ≥ 650 F to ≤ 200 F
	80 loss of load cycles, without immediate Turbine or Reactor Trip	≥ 15% of RATED THERMAL POWER to 0% of RATED THERMAL POWER.
	80 cycles of loss of flow in one reactor coolant loop	Loss of only one reactor coolant pump
	400 Reactor Trip cycles	100% to 0% of RATED THERMAL POWER
	40 leak tests	Pressurized to ≥ 2435 psig
	5 hydrostatic pressure tests	Pressurized to ≥ 3100 psig
Secondary Coolant System	6 Loss of Secondary Pressure	Loss of Secondary Pressure
	50 leak tests	Pressurized to ≥ 1085 psig
	10 hydrostatic pressure test	Pressurized to ≥ 1356 psig
	40 cycles of loss-of-offsite A.C. electrical power	Loss-of-offsite A.C. electrical ESF Electrical System

SECTION 6.0

ADMINISTRATIVE CONTROLS

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The Plant Manager - Nuclear shall be responsible for overall facility operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Plant Supervisor Nuclear (or during his absence from the control room, a designated individual) shall be responsible for the control room command function. A management directive to this effect, signed by the Group Vice President-Nuclear Energy shall be reissued to all station personnel on an annual basis.

6.2 ORGANIZATION

OFFSITE

6.2.1 The offsite organization for facility management and technical support shall be as shown in Figure 6.2-1.

FACILITY STAFF

6.2.2 The facility organization shall be as shown in Figure 6.2.2.

a. Each on-duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1;

b. At least one licensed Operator shall be in the control room when fuel is in the reactor. At least two licensed Operators shall be present in the Control Room during reactor startup, scheduled reactor shutdown and during recovery from reactor trips. In addition, while the unit is in MODE 1, 2, 3, or 4, at least one licensed Senior Operator shall be in the control room;

c. An individual qualified in radiation protection procedures shall be on site when fuel is in the reactor;

d. All CORE ALTERATIONS shall be directly supervised by either a licensed Senior Operator or licensed Senior Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation;

e. A site Fire Brigade of at least five members* shall be maintained on site at all times. The Fire Brigade shall not include the Shift Supervisor and two other members of the minimum shift crew necessary for safe shutdown of the unit and any personnel required for other essential functions during a fire emergency; and

* The fire Brigade composition may be less than the minimum requirements for a period of time not to exceed 2 hours, in order to accommodate unexpected absence, provided immediate action is taken to fill the required positions.

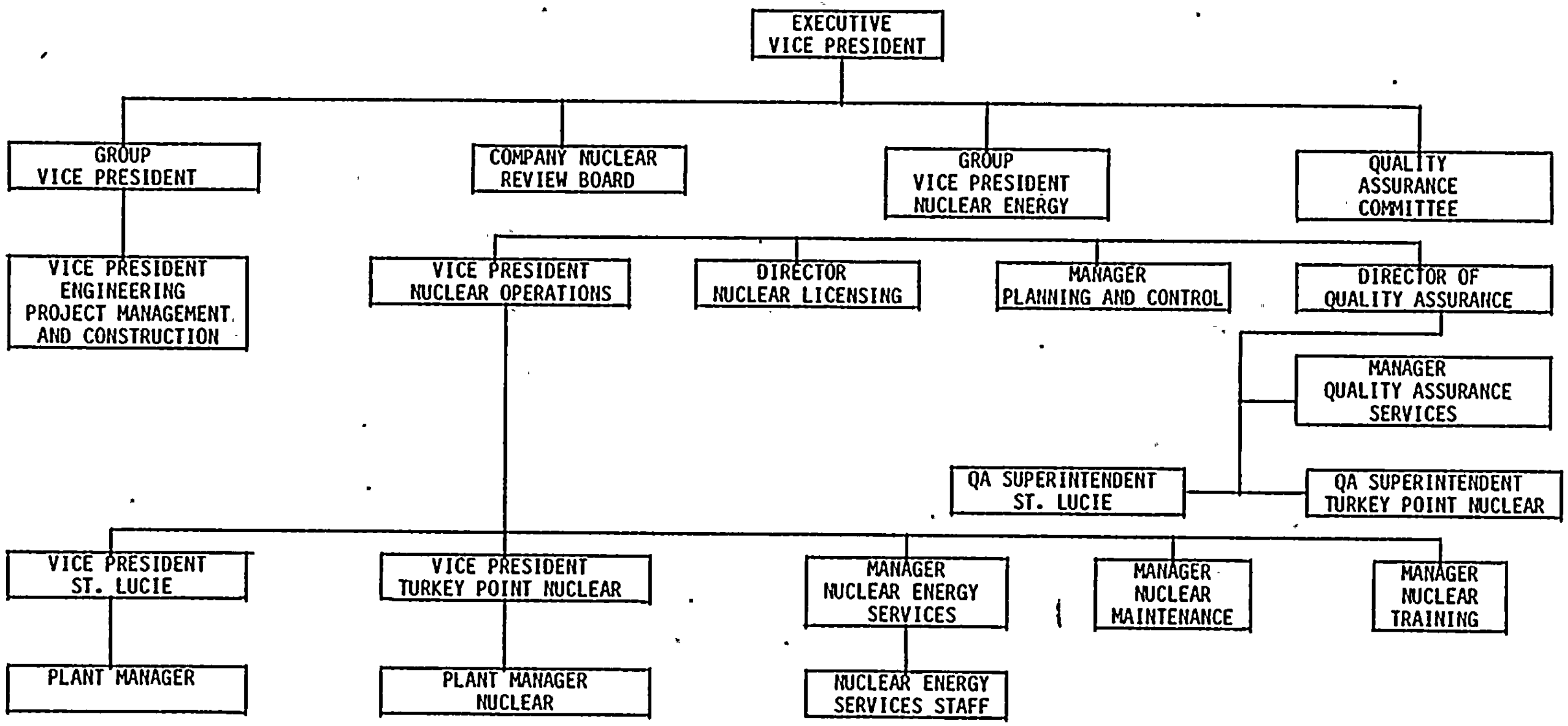
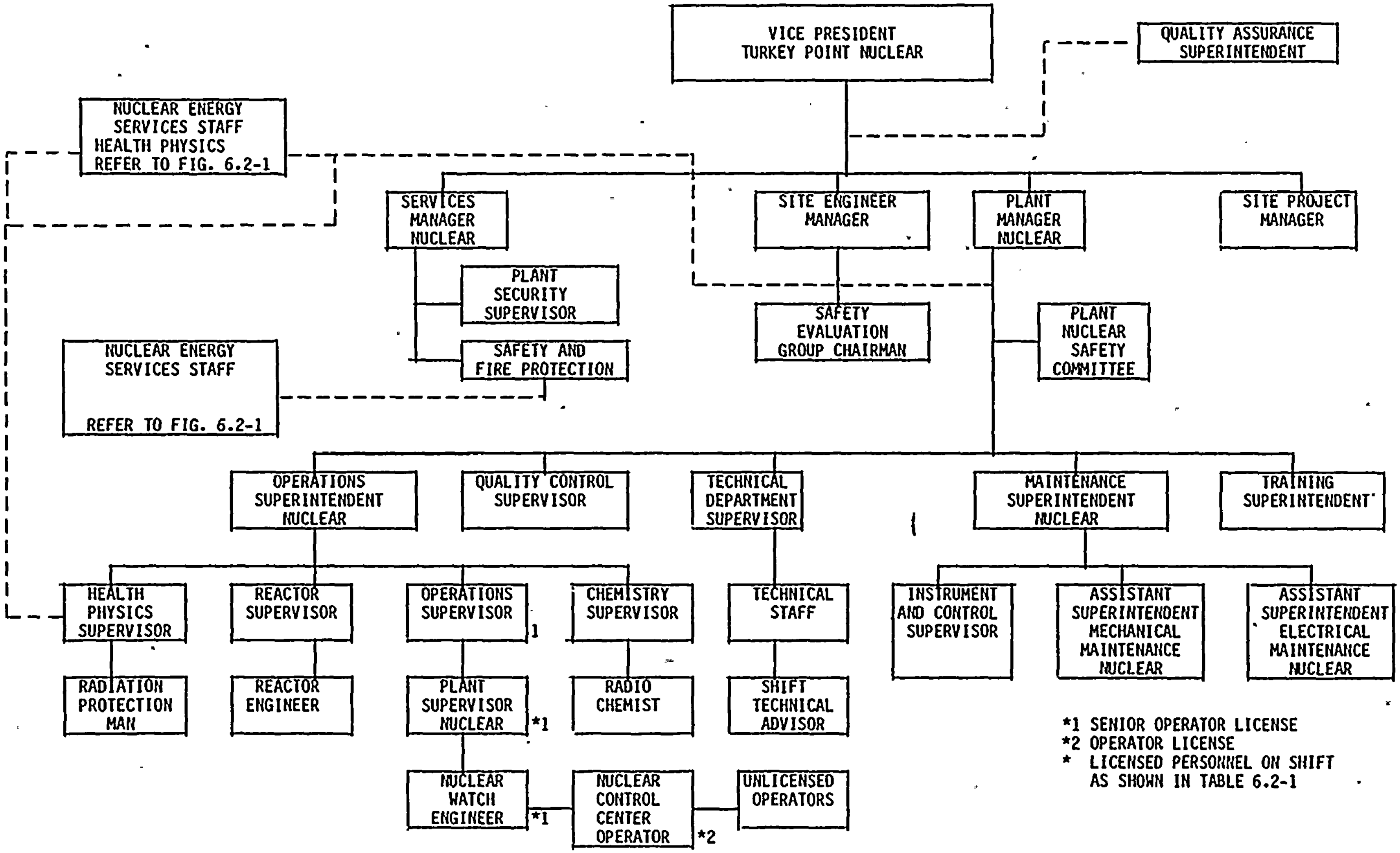


FIGURE 6.2-1
OFF-SITE ORGANIZATION FOR FACILITY
MANAGEMENT AND TECHNICAL SUPPORT



PLANT ORGANIZATION CHART
FIGURE 6.2-2

TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION

POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION		
	BOTH UNITS IN MODE 1, 2, 3, or 4	BOTH UNITS IN MODE 5 or 6 OR DEFUELED	ONE UNIT IN MODE 1, 2, 3, or 4 AND ONE UNIT IN MODE 5 or 6 OR DEFUELED
PSN	1	1	1
SRO	1	none**	1
RO	3*	2*	3*
AO	3*	3*	3*
STA	1***	none	1***

PSN - Plant Supervisor Nuclear with a Senior Operator license
SRO - Individual with a Senior Operator license
RO - Individual with an Operator license
AO - Auxiliary Operator
STA - Shift Technical Advisor

The shift crew composition may be one less than the minimum requirements of Table 6.2-1 for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

During any absence of the Plant Supervisor Nuclear from the control room while the unit is in MODE 1, 2, 3, or 4, an individual (other than STA) with a Senior Operator license shall be designated to assume the control room command function. During any absence of the Plant Supervisor Nuclear from the control room while both units are in MODE 5 or 6, an individual with Senior Operator license or Operator license shall be designated to assume the control room command function.

* At least one of the required individuals must be assigned to the designated position for each unit.

** At least one licensed Senior Operator or licensed Senior Operator Limited to Fuel Handling must be present during CORE ALTERATIONS on either unit, who has no other concurrent responsibilities.

*** The STA position shall be manned in MODES 1, 2, 3, and 4 unless the Plant Supervisor Nuclear or the individual with a Senior Operator license meets the qualifications for the STA as required by the NRC.

6.2 ORGANIZATION (Continued)

FACILITY STAFF (Continued)

f. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety-related functions (e.g., licensed Senior Operators, licensed Operators, health physicists, auxiliary operators, and key maintenance personnel).

Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel work a normal 8-hour day, 40-hour week while the unit is operating. However, in the event that unforseen problems require substantial amounts of overtime to be used, or during extended periods of shutdown for refueling, major maintenance, or major plant modification, on a temporary basis the following guidelines shall be followed:

1. An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time.

2. An individual should not be permitted to work more than 16 hours in any 24-hour period, nor more than 24 hours in any 48-hour period, nor more than 72 hours in any 7 day period, all excluding shift turnover time.

3. A break of at least 8 hours should be allowed between work periods, including shift turnover time.

4. Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

Any deviation from the above guidelines shall be authorized by the Plant Manager-Nuclear, or higher levels of management, in accordance with established procedures and with documentation of the basis for granting deviation. Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the Plant Manager-Nuclear or his designee to assure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.

6.2.3 SHIFT TECHNICAL ADVISOR

6.2.3.1 The Shift Technical Advisor shall provide advisory technical support to the Plant Supervisor Nuclear in the areas of thermal hydraulics, and plant analysis with regard to the safe operation of the units. The Shift Technical Advisor shall have a bachelor's degree or equivalent in a scientific or engineering discipline and shall have received specific training in the response and analysis of the units for transients and accidents, and in units design and layout, including the capabilities of instrumentation and controls in the control room.

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for the Health Physics Supervisor who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975. The licensed Operators and Senior Operators shall also meet or exceed the minimum qualifications of the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees to the extent stated in Florida Power and Light Company response dated August 1, 1980.

6.3.2 When the Health Physics Supervisor does not meet the above requirements, compensatory action shall be taken which the Plant Nuclear Safety Committee determines and the NRC office of Nuclear Reactor Regulation concurs that the action meets the intent of Specification 6.3.1.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the facility staff shall be maintained under the direction of the Training Supervisor and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and Appendix A of 10CFR Part 55 and shall include familiarization with relevant industry operational experience.

6.5 REVIEW AND AUDIT

6.5.1 PLANT NUCLEAR SAFETY COMMITTEE

FUNCTION

6.5.1.1 The PNSC shall function to advise the Plant Manager - Nuclear on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The PNSC shall be composed of the:

Chairman:	Plant Manager - Nuclear
Vice Chairman:	Operations Superintendent - Nuclear
Member:	Technical Department Supervisor
Member:	Maintenance Superintendent - Nuclear
Member:	Instrument and Control Supervisor
Member:	Reactor Supervisor
Member:	Health Physics Supervisor

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the PNSC Chairman to serve on a temporary basis; however, no more than two alternates shall participate in PNSC activities at any one time.

MEETING FREQUENCY

6.5.1.4 The PNSC shall meet at least once per calendar month and as convened by the PNSC Chairman or Vice Chairman.

QUORUM

6.5.1.5 The quorum of the PNSC necessary for the performance of the PNSC responsibility and authority provisions of these Technical Specifications shall consist of the Chairman or Vice Chairman and four members including alternates.

RESPONSIBILITIES

6.5.1.6 The PNSC shall be responsible for:

a. Review of: (1) all proposed procedures required by Specification 6.8.1a-g and changes thereto, (2) all proposed programs required by Specification 6.8.4 a through e and changes thereto, and (3) any other proposed procedures or changes thereto as determined by the Plant Manager-Nuclear to affect nuclear safety;

6.5 REVIEW AND AUDIT (Continued)

6.5.1 PLANT NUCLEAR SAFETY COMMITTEE

RESPONSIBILITIES (CONTINUED)

b. Review of all proposed tests and experiments that affect nuclear safety;

c. Review of all proposed changes to Appendix "A" Technical Specifications;

d. Review of all proposed changes or modifications to unit systems or equipment that affect nuclear safety;

e. Investigation of all violations of the Technical Specifications, including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence, to the Vice President of Nuclear Operations, Group Vice President of Nuclear Energy and to the Chairman of the Company Nuclear Review Board;

f. Review of all REPORTABLE EVENTS;

g. Review of facility operations to detect potential hazards to nuclear safety;

h. Performance of special reviews, investigations, or analyses and reports thereon as requested by the Plant Manager-Nuclear or the Chairman of the Company Nuclear Review Board.

i. Review of the Security Plan and implementing procedures and submittal of recommended changes to the Chairman of the Company Nuclear Review Board;

j. Review of the Emergency Plan and implementing procedures and submittal of recommended changes to the Chairman of the Company Nuclear Review Board;

k. Review of any accidental, unplanned, or uncontrolled radioactive release including the preparation of reports covering evaluation, recommendations, and disposition of the corrective action to prevent recurrence and the forwarding of these reports to the Group Vice President of Nuclear Energy and to the Chairman of the Company Nuclear Review Board; and

l. Review of changes to the PROCESS CONTROL PROGRAM, the OFFSITE DOSE CALCULATION MANUAL, and the Radwaste Treatment Systems.

6.5 REVIEW AND AUDIT (Continued)

6.5.1 PLANT NUCLEAR SAFETY COMMITTEE

RESPONSIBILITIES (Continued)

6.5.1.7 The PNSC shall:

a. Recommend in writing to the Plant Manager-Nuclear approval or disapproval of items considered under Specification 6.5.1.6a through d. prior to their implementation;

b. Render determination in writing with regard to whether or not each item considered under Specification 6.5.1.6a through e. constitutes an unreviewed safety question; and

c. Provide immediate written notification within 24 hours to the Vice President of Nuclear Operations and the Company Nuclear Review Board of disagreement between the PNSC and the Plant Manager-Nuclear; however, the Plant Manager-Nuclear shall have responsibility for resolution of such disagreements pursuant to Specification 6.1.1.

RECORDS

6.5.1.8 The PNSC shall maintain written minutes of each PNSC meeting that, at a minimum, document the results of all PNSC activities performed under the responsibility provisions of these Technical Specifications. Copies shall be provided to the Vice President of Nuclear Operations and the Company Nuclear Review Board.

6.5.2 COMPANY NUCLEAR REVIEW BOARD (CNRB)

FUNCTION

6.5.2.1 The CNRB shall function to provide independent review and audit of designated activities in the areas of:

a. Nuclear power plant operations,
b. Nuclear engineering,
c. Chemistry and radiochemistry,
d. Metallurgy,
e. Instrumentation and control,
f. Radiological safety,
g. Mechancical and electrical engineering, and
h. Quality assurance practices.

The CNRB shall report to and advise the Executive Vice President on those areas of responsibility specified in Specifications 6.5.2.7 and 6.5.2.8.

COMPOSITION

6.5.2.2 The CNRB shall be composed of the:

Chairman:	Group Vice President, Nuclear Energy
Member:	Group Vice President
Member:	Vice President, Engineering Project Management and Construction
Member:	Vice President, Nuclear Operations
Member:	Chief Engineer, Power Plant Engineering
Member:	Director, Quality Assurance
Member:	Director, Nuclear Licensing
Member:	Senior Project Manager, Power Plant Engineering
Member:	Manager, Nuclear Energy Services
Member:	Manager, Nuclear Fuels

ALTERNATES

6.5.2.3 All alternate members shall be appointed in writing by the CNRB Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in CNRB activities at any one time.

CONSULTANTS

6.5.2.4 Consultants shall be utilized as determined by the CNRB Chairman to provide expert advice to the CNRB.

MEETING FREQUENCY

6.5.2.5 The CNRB shall meet at least once per calendar quarter during the initial year of facility operation following fuel loading and at least once per 6 months thereafter.

QUORUM

6.5.2.6 The quorum of the CNRB necessary for the performance of the CNRB review and audit functions of these Technical Specifications shall consist of the Chairman or designated acting Chairman and at least four members including alternates. No more than a minority of the quorum shall have line responsibility for operation of the facility.

REVIEW

6.5.2.7 The CNRB shall be responsible for the review of:

a. The safety evaluations for: (1) changes to procedures, equipment, or systems; and (2) tests or experiments completed under the provision of 10CFR50.59, to verify that such actions did not constitute an unreviewed safety question;

b. Proposed changes to procedures, equipment, or systems which involve an unreviewed safety question as defined in 10CFR50.59;

c. Proposed tests or experiments which involve an unreviewed safety question as defined in 10CFR50.59;

d. Proposed changes to Technical Specifications or this Operating License;

e. Violations of applicable statutes, codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance;

f. Significant operating abnormalties or deviations from normal and expected performance of plant equipment that affect nuclear safety;

g. All REPORTABLE EVENTS;

h. All recognized indications of an unanticipated deficiency in some aspect of design or operation of structures, systems, or components that could affect nuclear safety; and

i. Reports and meeting minutes of the PNSC.

AUDITS

6.5.2.8 Audits of unit activities shall be performed under the cognizance of the CNRB. These audits shall encompass:

a. The conformance of facility operation to provisions contained within the Technical Specifications and applicable license conditions at least once per 12 months;

b. The performance, training, and qualifications of the entire facility staff at least once per 12 months;

c. The results of actions taken to correct deficiencies occurring in facility equipment, structures, systems, or method of operation that affect nuclear safety, at least once per 6 months;

AUDITS (Continued)

d. The performance of activities required by the Quality Assurance Program to meet the criteria of Appendix B, 10CFR part 50, at least once per 24 months;

e. The fire protection programatic controls including the implementing procedures at least once per 24 months by qualified licensee QA personnel;

f. The fire protection equipment and program implementation at least once per 12 months utilizing either a qualified offsite licensee fire protection engineer or an outside independent fire protection consultant. An outside independent fire protection consultant shall be used at least every third year;

g. The Radiological Environmental Monitoring Program and the results thereof at least once per 12 months;

h. The OFFSITE DOSE CALCULATION MANUAL and implementing procedures at least once per 24 months;

i. The PROCESS CONTROL PROGRAM and implementing procedures for processing and packaging of radioactive wastes at least once per 24 months;

j. The performance of activities required by the Quality Control Program for effluent and environmental monitoring at least once per 12 months;

k. The Emergency Plans and implementing procedures at least once per year;

l. The Security Plans and implementing procedures at least once per year; and

m. Any other area of facility operation considered appropriate by the CNRB or the Executive Vice President.

RECORDS

6.5.2.9 Records of CNRB activities shall be prepared, approved, and distributed as indicated below:

a. Minutes of each CNRB meeting shall be prepared, approved, and forwarded to the Executive Vice President within 14 days following each meeting;

b. Reports of reviews encompassed by Specification 6.5.2.7 shall be prepared, approved, and forwarded to the Executive Vice President within 14 days following completion of the review; and

c. Audit reports encompassed by Specification 6.5.2.8 shall be forwarded to the Executive Vice President and to the management positions responsible for the areas audited within 30 days after completion of the audit by the auditing organization.

6.6 REPORTABLE EVENT ACTION

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified and a report submitted pursuant to the requirements of Section 50.73 to 10CFR Part 50, and

b. Each REPORTABLE EVENT shall be reviewed by the PNSC, and the results of this review and the report shall be submitted to the CNRB, the Vice President of Nuclear Operations, and the Group Vice President of Nuclear Energy.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The NRC Operations Center shall be notified by Emergency Notification System as soon as possible and in all cases within 1 hour. The Group Vice President of Nuclear Energy, Vice President of Nuclear Operations, and the CNRB shall be notified immediately;

b. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the PNSC. This report shall describe: (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems, or structures, and (3) corrective action taken to prevent reoccurrence;

c. The Safety Limit Violation Report shall be submitted to the Commission, the CNRB, the Group Vice President of Nuclear Energy and the Vice President of Nuclear Operations within 10 days of the violation; and

d. Operation of the unit shall not be resumed until authorized by the Commission.

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures and administrative policies shall be established, implemented, and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978, Sections 5.1 and 5.3 of ANSI N18.7-1972, and the Facility Fire Protection Program;

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and supplement 1 to NUREG-0737 as stated in Generic Letter No. 82-33;

c. Security Plan implementation;

d. Emergency Plan implementation;

e. PROCESS CONTROL PROGRAM implementation;

f. OFFSITE DOSE CALCULATION MANUAL implementation;

g. Quality Control Program for effluent monitoring using the guidance in Regulatory Guide 1.21, Revision 1, June 1974; and

h. Quality Control Program for environmental monitoring using the guidance in Regulatory Guide 4.1, Revision 1, April 1975.

6.8 PROCEDURES AND PROGRAMS

6.8.2 Each procedure of Specification 6.8.1a through g, and changes thereto, shall be reviewed by the PNSC and shall be approved by the Plant Manager-Nuclear prior to implementation and reviewed periodically as set forth in administrative procedures.

6.8.3 Temporary changes to procedures of Specification 6.8.1a through g may be made provided:

a. The intent of the original procedure is not altered;

b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Operator license on the unit affected; and

c. The change is documented, reviewed by the PNSC, and approved by the Plant Manager-Nuclear within 14 days of implementation.

6.8.4 The following programs shall be established, implemented, and maintained:

a. Primary Coolant Sources Outside Containment

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include safety injection, chemical volume control, and containment spray system. The program shall include the following:

1) Preventive maintenance and periodic visual inspection requirements, and

2) Integrated leak test requirements for each system at refueling cycle intervals or less.

b. In-Plant Radiation Monitoring

A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

1) Training of personnel,

2) Procedures for monitoring, and

3) Provisions for maintenance of sampling and analysis equipment.

6.8 PROCEDURES AND PROGRAMS

c. Secondary Water Chemistry

A program for monitoring of secondary water chemistry to inhibit steam generator tube degradation. This program shall include:

1) Identification of a sampling schedule for the critical variables and control points for these variables,

2) Identification of the procedures used to measure the values of the critical variables,

3) Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in-leakage,

4) Procedures for the recording and management of data,

5) Procedures defining corrective actions for all off-control point chemistry conditions, and

6) A procedure identifying: (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective action.

d. Backup Method for Determining Subcooling Margin

A program which will ensure the capability to accurately monitor the Reactor Coolant System subcooling margin. This program shall include the following:

1) Training of personnel, and

2) Procedures for monitoring.

e. Post-Accident Sampling

A program which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

1) Training of personnel,

2) Procedures for sampling and analysis, and

3) Provisions for maintenance of sampling and analysis equipment.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Regional Administrator of the Regional Office of the NRC unless otherwise noted.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following: (1) receipt of an Operating License, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design and/or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the unit.

The Startup Report shall address each of the tests identified in the Final Safety Analysis Report and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

Startup Reports shall be submitted within: (1) 90 days following completion of the Startup Test Program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of Startup Test Program, and resumption or commencement of commercial operation), supplementary reports shall be submitted at least every 3 months until all three events have been completed.

ANNUAL REPORTS*

6.9.1.2 Annual Reports covering the activities of the facility as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

* A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

6.9 REPORTING REQUIREMENTS (Continued)

Reports required on an annual basis shall include:

a. A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man-rem exposure according to work and job functions** (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance, waste processing, and refueling). The dose assignments to various duty functions may be estimated based on pocket dosimeter, thermoluminescent dosimeter (TLD), or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole-body dose received from external sources should be assigned to specific major work functions;

b. The results of specific activity analyses in which the primary coolant exceeded the limits of Specification 3.4.8. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded (in graphic and tabular format); (2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration (μCi/gm) and one other radioiodine isotope concentration (μCi/gm) as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT*

6.9.1.3 Routine Annual Radiological Environmental Operating Reports covering the operation of the facility during the previous calendar year shall be submitted prior to May 1 of each year.

The Annual Radiological Environmental Operating Reports shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activities for the report period including a comparison with preoperational studies, with operational controls, as appropriate, and with previous environmental surveillance reports, and an

* A single submittal may be made for a multiple unit station.
** This tabulation supplements the requirements of 20.407 of 10 CFR Part 20.

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6.9 REPORTING REQUIREMENTS (Continued)

assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of the Land Use Census required by Specification 3.12.2.

The Annual Radiological Environmental Operating Reports shall include the results of analysis of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the Offsite Dose Calculation Manual, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November, 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

The reports shall also include the following: a summary description of the Radiological Environmental Monitoring Program; at least two legible maps* covering all sampling locations keyed to a table giving distances and directions from the centerline of the reactor; the results of licensee participation in the Interlaboratory Comparison Program and the corrective action taken if the specified program is not being performed as required by Specification 3.12.3; reason for not conducting the Radiological Environmental Monitoring Program as required by Specification 3.12.1, and discussion of all deviations from the sampling schedule of Table 3.12-1; discussion of environmental sample measurements that exceed the reporting levels of Table 3.12-2 but are not the result of plant effluents, pursuant to ACTION b. of Specification 3.12.1; and discussion of all analyses in which the LLD required by Table 4.12-1 was not achievable.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT**

6.9.1.4 Routine Semiannual Radioactive Effluent Release Reports covering the operation of the facility during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year.

* One map shall cover stations near the SITE BOUNDARY; a second shall include the more distant stations.

** A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

6.9 REPORTING REQUIREMENTS (Continued)

The Semiannual Radioactive Effluent Release Reports shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the facility as outlined in Regulatory Guide 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," Revision 1, June 1974, with data summarized on a quarterly basis following the format of Appendix B thereof.

For each class of solid waste (as defined by 10 CFR Part 61) shipped offsite during the reporting period the following information shall be provided: type of solid wastes (e.g., dewatered spent resin, compacted dry waste, evaporator bottoms), type of container (e.g., LSA, Type A, Type B, Large Quantity) SOLIDIFICATION agent or absorbent (e.g., cement, urea formaldehyde), volume, total curie quantity, and principal radionuclides.

The Semiannual Radioactive Effluent Release Report to be submitted within 60 days after January 1 of each year shall include an annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the form of an hour-by-hour listing on magnetic tape of wind speed, wind direction, atmospheric stability, and precipitation (if measured), or in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability.* This same report shall include an assessment of the radiation doses due to the radioactive liquid and gaseous effluents released from the unit or station during the previous calendar year. This same report shall also include an assessment of the radiation doses from radioactive liquid and gaseous effluents to MEMBERS OF THE PUBLIC due to their activities inside the SITE BOUNDARY (Figure 5.1-1) during the report period. All assumptions used in making these assessments, i.e., specific activity, exposure time, and location, shall be included in these reports. The meteorological conditions concurrent with the time of release of radioactive materials in gaseous effluents, as determined by sampling frequency and measurement, shall be used for determining the gaseous pathway doses. The assessment of radiation doses shall be performed in accordance with the methodology and parameters in the OFFSITE DOSE CALCULATION MANUAL (ODCM).

The Semiannual Radioactive Effluent Release Report to be submitted within 60 days after January 1 of each year shall also include an assessment of radiation doses to the likely most exposed MEMBER OF THE PUBLIC from reactor releases from the previous calendar year. Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in Regulatory Guide 1.109, March 1976.

* In lieu of submission with the Semiannual Radioactive Effluent Release Report, the Licensee has the option of retaining this summary of required meteorological data on site in a file that shall be provided to the NRC upon request.

6.9 REPORTING REQUIREMENTS (Continued)

The Semiannual Radioactive Effluent Release Reports shall include a list and description of unplanned releases from the site to UNRESTRICTED AREAS of radioactive materials in gaseous and liquid effluents made during the reporting period.

The Semiannual Radioactive Effluent Release Reports shall include any changes made during the reporting period to the PROCESS CONTROL PROGRAM and the ODCM, pursuant to Specifications 6.13 and 6.14, respectively, as well as any major change to Liquid, Gaseous, or Solid Radwaste Treatment Systems pursuant to Specification 6.15. It shall also include a listing of new locations for dose calculations and/or environmental monitoring identified by the Land Use Census pursuant to Specification 3.12.2.

The Semiannual Radioactive Effluent Release Reports shall also include the following: an explanation as to why the inoperability of liquid or gaseous effluent monitoring instrumentation was not corrected within the time specified in Specification 3.3.3.6 or 3.3.3.7, respectively; and description of the events leading to gas storage tanks exceeding the limits of Specification 3.11.2.6.

MONTHLY OPERATING REPORTS

6.9.1.5 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the PORV's or safety valves, shall be submitted on a monthly basis to the Director, Office of Resource Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Administrator of the Regional Office of the NRC to arrive no later than the 15th of each month following the calendar month covered by the report.

PEAKING FACTOR LIMIT REPORT

6.9.1.6 The $W(z)_{BL}$ function for Base Load operation, $F_Z(Z)$ for Radial Burndown and the value for $[F_Q]^P$ needed to compute P_T (augmented surveillance power fraction, if required) shall be provided to the Director, Nuclear Reactor Regulation, Attention: Chief, Core Performance Branch, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555 at least 30 days prior to cycle initial criticality. In the event that these values would be submitted at some other time during core life, it will be submitted 30 days prior to the date the values would become effective unless otherwise exempted by the Commission. Any information needed to support $W(z)_{BL}$, $F_Z(Z)$ and P_T will be by request from the NRC and need not be included in this report.

SPECIAL REPORTING REQUIREMENTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the Regional Office of the NRC within the time period specified for each report as contained in specifications within Section 3.0. In addition, reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification where appropriate.

Twenty copies of the following reports should be sent to the Director, Nuclear Reactor Regulation.

(1) In-service inspection, reference 4.4.5 and 4.0.5.

(2) Tendon surveillance, reference 3.6.1.6.

6.10 RECORD RETENTION

6.10.1 In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.10.2 The following records shall be retained for at least 5 years:

a. Records and logs of facility/operation covering time interval at each power level;

b. Records and logs of principal maintenance activities, inspections, repair, and replacement of principal items of equipment related to nuclear safety;

c. ALL REPORTABLE EVENTS;

d. Records of surveillance activities, inspections, and calibrations required by these Technical Specifications;

e. Records of changes made to the procedures required by Specification 6.8.1;

f. Records of radioactive shipments;

g. Records of sealed source and fission detector leak tests and results; and

h. Records of annual physical inventory of all sealed source material of record.

6.10 RECORD RETENTION (Continued)

6.10.3 The following records shall be retained for the duration of the facility Operating License:

a. Records and drawing changes reflecting facility design modifications made to systems and equipment described in the Final Safety Analysis Report;

b. Records of new and irradiated fuel inventory, fuel transfers, and assembly burnup histories;

c. Records of facility radiation and contamination surveys;

d. Records of radiation exposure for all individuals entering radiation control areas;

e. Records of gaseous and liquid radioactive material released to the environs;

f. Records of transient or operational cycles for those components identified in Table 5.7-1;

g. Records of reactor tests and experiments;

h. Records of training and qualification for current members of the facility staff;

i. Records of inservice inspections performed pursuant to these Technical Specifications;

j. Records of quality assurance activities required for the duration of facility operating license by the Quality Assurance Manual except as listed in Specification 6.10.2;

k. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59;

l. Records of meetings of the PNSC and the CNRB;

m. Records of the service lives of all snubbers required by Specification 3.7.8 including the date at which the service life commences and associated installation and maintenance records;

n. Records of secondary water sampling and water quality; and

o. Annual Radiological Environmental Monitoring Reports; and records of analyses transmitted to the licensee which are used to prepare the Annual Radiological Environmental Monitoring Reports.

6.11 RADIATION PROTECTION PROGRAM

6.11.1 Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained, and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

6.12.1 Pursuant to Paragraph 20.203(c)(5) of 10 CFR Part 20, in lieu of the "control device" or "alarm signal" required by Paragraph 20.203(c), each high radiation area, as defined in 10 CFR Part 20, in which the intensity of radiation is equal to or less than 1000 mR/hr at 45 cm (18 in.) from the radiation source or from any surface which the radiation penetrates shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP). Individuals qualified in radiation protection procedures (e.g., Health Physics Technician) or personnel continuously escorted by such individuals may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates equal to or less than 1000 mR/h, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area; or

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel have been made knowledgeable of them; or

c. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the facility Health Physics Shift Supervisor in the RWP.

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6.12.2 In addition to the requirements of Specification 6.12.1, areas accessible to personnel with radiation levels greater than 1000 mR/hr at 45 cm (18 in.) from the radiation source or from any surface which the radiation penetrates shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the adminstrative control of the shift Foreman on duty and/or health physics supervision. Doors shall remain locked except during periods of access by personnel under an approved RWP which shall specify the dose rate levels in the immediate work areas and the maximum allowable stay time for individuals in that area. In lieu of the stay time specification of the RWP, direct or remote (such as closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the acitivities being performed within the area.

For individual high radiation areas accessible to personnel with radiation levels of greater than 1000 mR/hr at 45 cm(18 in.) that are located within large areas, such as PWR containment, where no enclosure exists for purposes of locking, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be roped off, conspicuously posted, and a flashing light shall be activated as a warning device.

6.13 PROCESS CONTROL PROGRAM (PCP)

6.13.1 The PCP shall be approved by the Commission prior to implementation.

6.13.2 Licensee-initiated changes to the PCP:

a. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:

(1) Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information;

(2) A determination that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and

(3) Documentation of the fact that the change has been reviewed and found acceptable by the PNSC.

b. Shall become effective upon review and acceptance by the PNSC.

[illegible]

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.14.1 The ODCM shall be approved by the Commission prior to implementation.

6.14.2 Licensee-initiated changes to the ODCM:

a. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made effective. This submittal shall contain:

(1) Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the ODCM to be changed with each page numbered, dated and containing the revision number, together with appropriate analyses or evaluations justifying the change(s);

(2) A determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations; and

(3) Documentation of the fact that the change has been reviewed and found acceptable by the PNSC.

b. Shall become effective upon review and acceptance by the PNSC.

6.15 MAJOR CHANGES TO LIQUID, GASEOUS, AND SOLID RADWASTE TREATMENT SYSTEMS*

6.15.1 Licensee-initiated major changes to the Radwaste Treatment Systems (liquid, gaseous, and solid).

a. Shall be reported to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the evaluation was reviewed by the PNSC. The discussion of each change shall contain:

(1) A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR 50.59;

(2) Sufficient detailed information to totally support the reason for the change without benefit of additonal or supplemental information;

(3) A detailed description of the equipment, components, and processes involved and the interfaces with other plant systems;

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* Licensees may choose to submit the information called for in this Specification as part of the annual FSAR update.

6.15 MAJOR CHANGES TO LIQUID, GASEOUS, AND SOLID RADWASTE TREATMENT SYSTEMS*

(4) An evaluation of the change, which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments thereto:

(5) An evaluation of the change, which shows the expected maximum exposures to a MEMBER OF THE PUBLIC IN the UNRESTRICTED AREA and to the general population that differ from those previously estimated in the license application and amendments thereto;

(6) A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste to the actual releases for the period prior to when the change is to be made;

(7) An estimate of the exposure to plant operating personnel as a result of the change; and

(8) Documentation of the fact that the change was reviewed and found acceptable by the PNSC.

b. Shall become effective upon review and acceptance by the PNSC.

* Licensees may choose to submit the information called for in this Specification as part of the annual FSAR update.