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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

RxC

August 1, 2003

MEMORANDUM TO: John T. Larkins, Executive Director
Advisory Committee on Reactor Safeguards and
Advisory Committee on Nuclear Waste

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ACRS/ACNW
US NRC
SEP - 2 2003

FROM: Ledyard B. Marsh, Director
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

SUBJECT: REVIEW STANDARD FOR EXTENDED POWER UPRATES

Attached is Review Standard RS-001, "Review Standard for Extended Power Uprates." This Review Standard is being provided to support the August meeting of the Advisory Committee on Reactor Safeguards (ACRS) Thermal-Hydraulic Phenomena Subcommittee and the September meeting of the ACRS Full Committee. The Review Standard was discussed with the ACRS in July and December 2002 and was issued in draft form for interim use and public comment in December 2002. A summary of the public comments received by the staff on Draft RS-001 is provided in Attachment 2. A summary of comments the staff received from the ACRS during review of past extended power uprate applications is provided in Attachment 3.

In parallel with developing RS-001, the staff was also updating Standard Review Plan (SRP) Sections 13.2.1, "Reactor Operator Training," 13.2.2, "Training for Nonlicensed Plant Staff," 13.5.2.1, "Operating and Emergency Operating Procedures," and SRP Chapter 18.0, "Human Factors Engineering." These SRP sections and chapter were also issued for public comment in December 2002. The staff received no public comments on SRP Sections 13.2.1 and 13.2.2, minor comments on SRP Section 13.5.2.1, and several comments on the reference to NUREG-1764 in SRP Chapter 18.0. The staff is attaching SRP Sections 13.2.1, 13.2.2 and 13.5.2.1 for ACRS review in parallel with RS-001. SRP Chapter 18.0 is attached for completeness. The staff is not currently requesting ACRS review of this chapter. The staff will request a separate ACRS review of this chapter once public comments have been addressed.

The staff also developed SRP Section 14.2.1, "Generic Guidelines for Extended Power Uprate Testing Programs," in parallel with RS-001 and issued it for public comment in December 2002. The staff received no comments on this SRP section. SRP Section 14.2.1 is specific to extended power uprates and is being forwarded to the ACRS for review in parallel with RS-001.

Attachments:
1. RS-001, "Review Standard for Extended Power Uprates"
2. Public Comments on Draft RS-001
3. ACRS Comments on Past Extended Power Uprate Reviews
4. SRP Section 13.2.1, "Reactor Operator Training"
5. SRP Section 13.2.2, "Training for Nonlicensed Plant Staff"
6. SRP Section 13.5.2.1, "Operating and Emergency Operating Procedures"
7. SRP Chapter 18.0, "Human Factors Engineering"
8. SRP Section 14.2.1, "Generic Guidelines for Extended Power Uprate Testing Programs"

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OFFICE OF NUCLEAR REACTOR REGULATION

REVIEW STANDARD FOR EXTENDED POWER UPRATES

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RS-001, Revision 0
[redacted]

ATTACHMENT 1

RS-001, “Review Standard for Extended Power Uprates”

RS-001, "Review Standard for Extended Power Uprates"

RS-001 CHANGE HISTORY			
Date	**Description of Changes**	**Method Used to Announce & Distribute**	**Training**
12/2002	Initial issuance for interim use and public comment	• *Federal Register* • Power Uprate Web site • ADAMS	[TBD]

[redacted]	Issuance of RS-001, Revision 0
	• Revised the Purpose section to add paragraphs 3 thru 5 to reflect changes resulting from public comments
	• Reformatted matrices in Section 2 and SE inserts in Section 3 to reflect NRR reorganization
	• Revised Section 2 to add specific criteria for independent calculations for Containment Review Considerations
	• Revised the matrix for Mechanical and Civil Engineering to add a note to highlight experience with dryer failures at Quad Cities 2 and identify focus of staff review in relation to this experience
	• Revised the matrix for Reactor Systems to:
	  • delete the reference to the ISCOR computer code and spectrum of breaks analyzed in the note on BWR reviews
	  • delete the bullet regarding hot leg streaming
	  • deleted the note regarding overfill analyses for SGTR
	  • delete reference to Item II.K.3.5 of NUREG-0737 in the note on LOCA reviews
	  • combine and reformat the notes on ATWS reviews
	• Added two notes to the matrix in Section 2 for Health Physics to identify obsolete guidance
	• Revised the regulatory evaluation sections of the SE inserts for Health Physics in Section 3 to add the statement that the NRC also considers the effects of the proposed EPU on plant effluent levels and any effect this increase may have on radiation doses at the site boundary
	• Revised the regulatory evaluation sections of the SE inserts for Human Performance in Section 3 to add a reference to GL 82-33
	• Revised the conclusion sections of the SE inserts for Power Ascension and Testing Plan in Section 3 to make them consistent with the wording in proposed SRP Section 14.2.1
	• Made miscellaneous editorial changes
		• *Federal Register*
		• Power Uprate Web site
		• ADAMS	

TABLE OF CONTENTS

PURPOSE

The purpose of this review standard is to provide guidance for the Nuclear Regulatory Commission (NRC) staff's review of extended power uprate (EPU) applications to enhance consistency, quality, and completeness of reviews.

This review standard also informs licensees of the guidance documents the staff uses when reviewing EPU applications. These documents provide acceptance criteria for the areas of review. This should allow licensees to prepare EPU applications that are complete with respect to the areas that are within the staff's scope of review. To further improve the efficiency of the staff's review of EPU applications, licensees are encouraged to provide, with their EPU applications, markups of the matrices in Section 2.1 of this review standard to identify any differences between the information in the review standard and the licensing bases of their plants.

Use of this review standard should not undermine the NRC's longstanding topical report review and approval process. If a licensee references the NRC approved topical report for an area covered by this review standard, the topical report will be considered as part of the review.

In addition, this review standard should not be used as the sole reason for imposing any new licensing or design requirements. If the staff identifies the need for imposing new requirements during its review of an EPU application, the NRC's backfit rule (10 CFR 50.109) should be invoked and the associated process followed.

In addition to this review standard, the NRC maintains a Web site on power uprates at http://www.nrc.gov/reactors/operating/licensing/power-uprates.html. Some of the material on this Web site includes:

- the status of completed, ongoing, and expected power uprate reviews
- general guidance related to power uprates
- references to publicly available correspondence related to reviews of recently completed power uprates (including licensees' responses to NRC staff requests for additional information, as well as NRC staff safety evaluations)

BACKGROUND

Facility operating licenses and technical specifications specify the maximum power level at which commercial nuclear power plants may be operated. NRC approval is required for any changes to facility operating licenses or technical specifications. The process for making changes to facility operating licenses and technical specifications is governed by Title 10 of the *Code of Federal Regulations*, Part 50.

The process of increasing the licensed power level at a commercial nuclear power plant is called a "power uprate." Power uprates are categorized based on the magnitude of the power increase and the methods used to achieve the increase. Measurement uncertainty recapture power uprates result in power level increases that are less than 2 percent and are achieved by implementing enhanced techniques for calculating reactor power. Stretch power uprates typically result in power level increases that are up to 7 percent and do not generally involve major plant modifications. EPUs result in power level increases that are greater than stretch power uprates and usually require significant modifications to major plant equipment. The NRC has approved EPUs for increases as high as 20 percent. This review standard is applicable to EPUs.

This review standard establishes standardized review guidance and acceptance criteria for the staff's reviews of EPU applications to enhance consistency, quality, and completeness of reviews. It serves as a tool for the staff's use when processing EPU applications in that it provides detailed references to various NRC documents containing information related to the specific areas of review.

This review standard also informs licensees of the guidance documents the staff will use when reviewing EPU applications. This will help licensees prepare EPU applications that address those topics required for a complete application. By addressing the areas in the review standard, a licensee could prepare and submit a more complete application and thus minimize the staff's need for requests for additional information (RAIs). This would improve the efficiency of the staff's reviews.

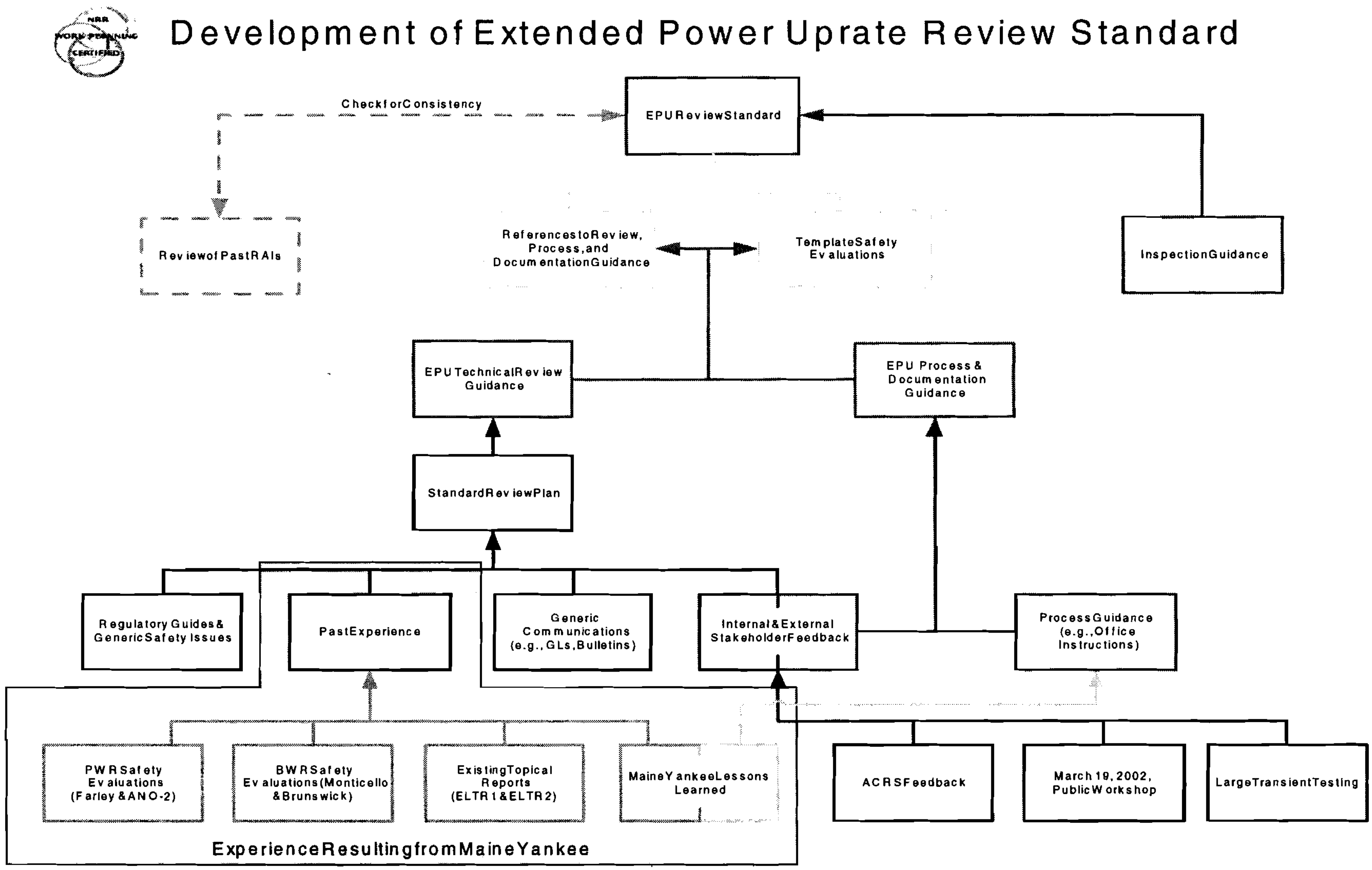
The development of this review standard included an evaluation of the NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (SRP), to determine the applicability and adequacy of the various SRP sections to the review of EPU applications and development/revision of guidance, as necessary. During this evaluation, the staff considered the versions of the SRP sections identified in the matrices in Section 2 of this review standard. To determine the need for guidance beyond that in the SRP, the staff reviewed: (1) safety evaluations for previously approved power uprates, (2) previously approved topical reports for EPUs, (3) various reports related to Maine Yankee Lessons Learned, and (4) generic communications. The staff also considered feedback from internal and external stakeholders. In addition, the staff reviewed RAIs issued for recent EPU applications to ensure that the review standard adequately addresses areas where repeat RAIs have been issued.

The staff reviewed NRC procedural guidance documents to identify those applicable to processing EPU applications. The review of these documents also included consideration of the recommendations in various reports related to Maine Yankee Lessons Learned and the feedback received from internal and external stakeholders.

Figure 1 provides a graphical representation of the development of the review standard.

FIGURE 1

Development of Extended Power Uprate Review Standard



GUIDANCE

This review standard provides guidance for

- processing EPU applications (Section 1)
- performing technical reviews (Section 2)
- preparing safety evaluations to document the reviews (Section 3)

This review standard also includes a reference to the NRC's Inspection Manual, which provides guidance for conducting inspections related to the implementation of power uprates (Section 4).

SECTION 1

PROCEDURAL GUIDANCE

1.1 Processing Extended Power Uprate Applications

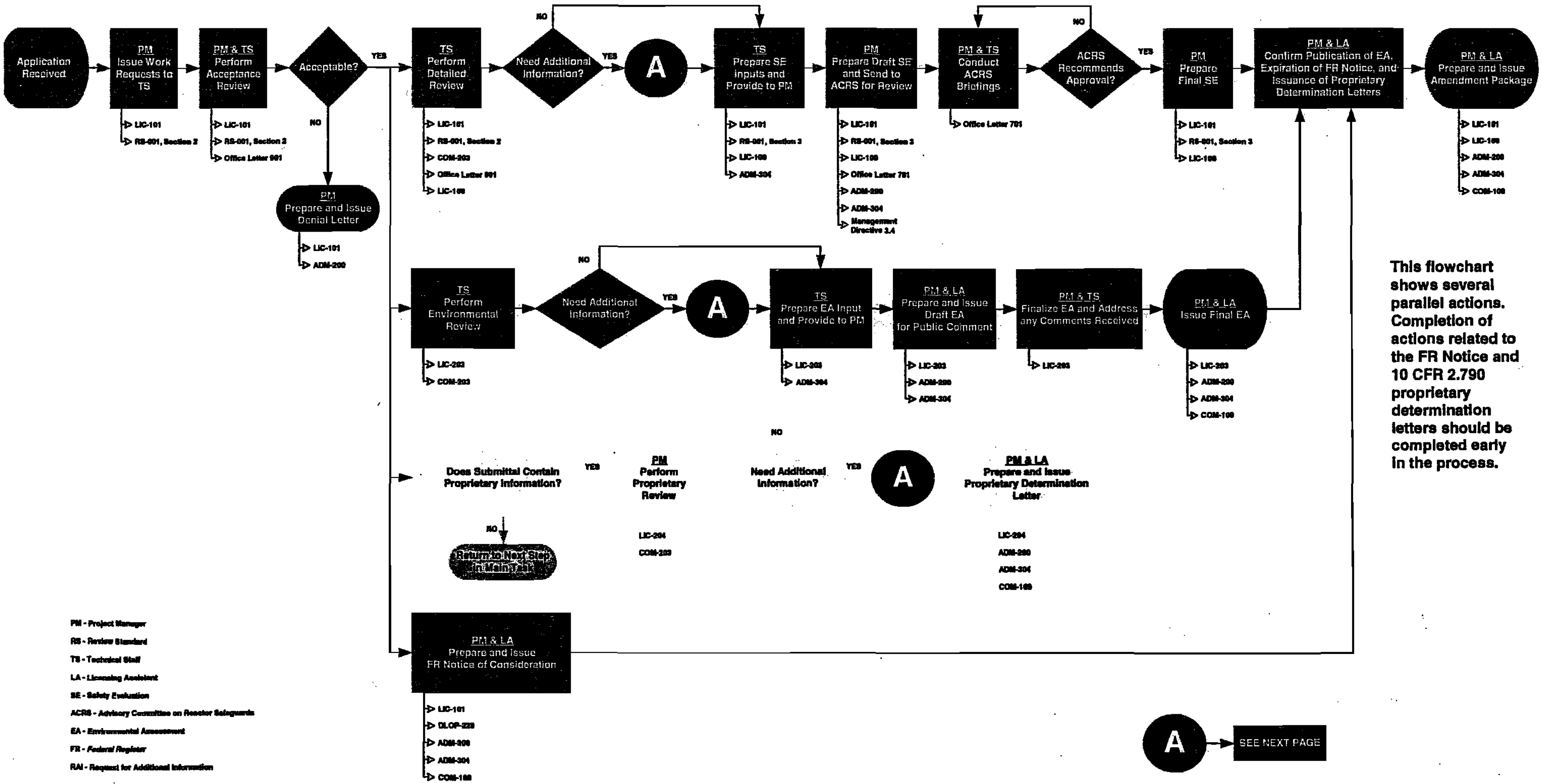
The process flow chart (Figure 1.1-1) identifies each step involved in processing an EPU application. The flow chart also identifies the responsible individual/organization and applicable procedures for completing each step. The staff should use the flow chart and referenced guidance documents when processing EPU applications.

Processing an EPU application involves, but is not limited to:

- performing an acceptance review
- issuing a *Federal Register* notice (without making a proposed no significant hazards consideration determination)
- performing a detailed technical review
- conducting ACRS briefings
- issuing draft and final environmental assessments
- making proprietary determinations, as necessary

The cognizant licensing Project Manager is responsible for coordinating the staff's review and ensuring that it is conducted in accordance with the process defined herein.

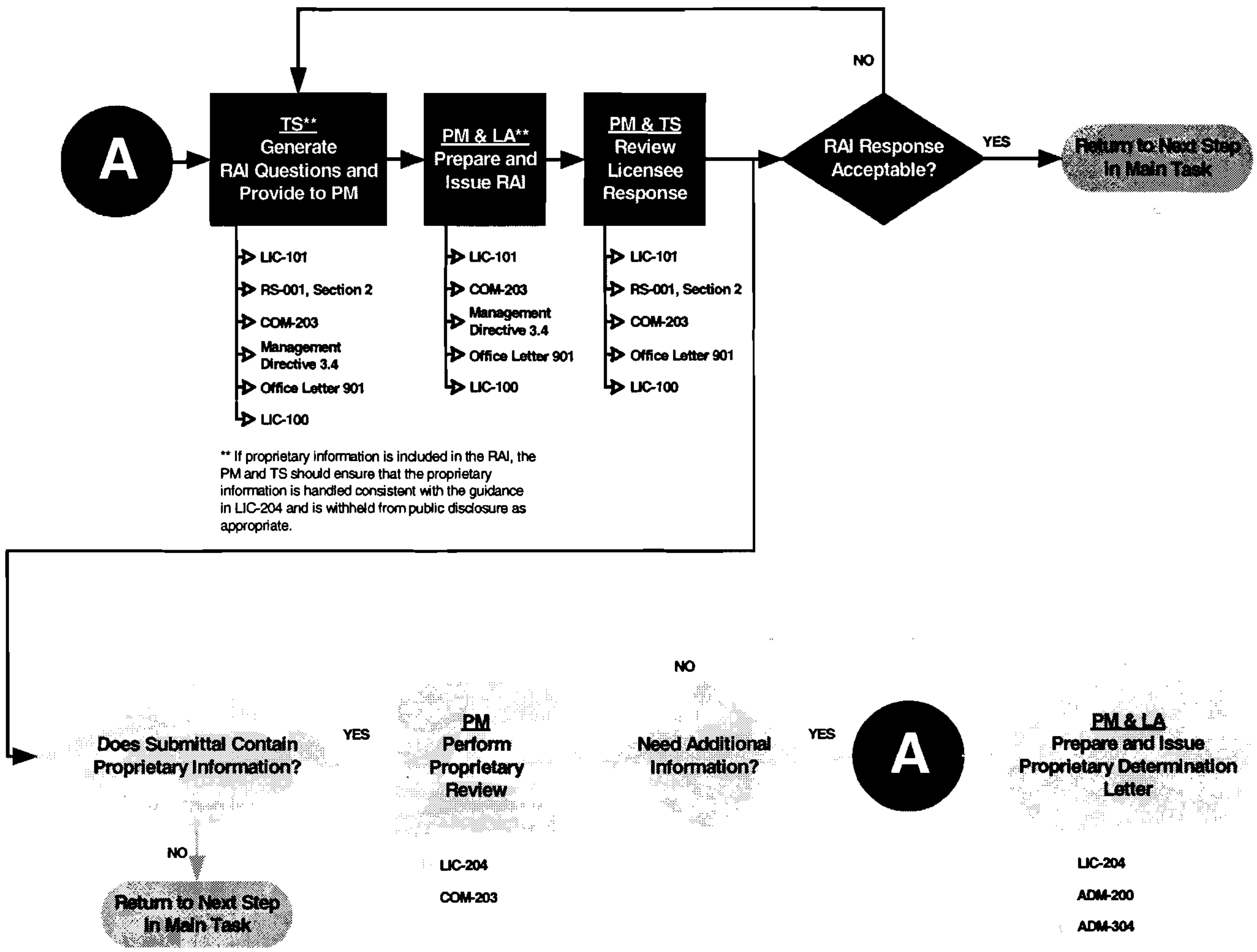
Figure 1.1-1 EPU Process Flow Chart



This flowchart shows several parallel actions. Completion of actions related to the FR Notice and 10 CFR 2.790 proprietary determination letters should be completed early in the process.

PM - Project Manager
RS - Review Standard
TS - Technical Staff
LA - Licensing Assistant
SE - Safety Evaluation
ACRS - Advisory Committee on Reactor Safeguards
EA - Environmental Assessment
FR - *Federal Register*
RAI - Request for Additional Information

Figure 1.1-1 EPU Process Flow Chart
continued



PM - Project Manager

RS - Review Standard

TS - Technical Staff

LA - Licensing Assistant

SE - Safety Evaluation

ACRS - Advisory Committee on Reactor Safeguards

EA - Environmental Assessment

FR - *Federal Register*

RAI - Request for Additional Information

SECTION 2

TECHNICAL REVIEW GUIDANCE

2.1 Reviewing Extended Power Uprate Applications

This section defines the scope of technical review for EPU applications and identifies the guidance to be used when performing technical reviews of such applications.

Matrices 1 thru 11 of this section identify: (1) the technical areas to be reviewed, (2) the technical branches within the Office of Nuclear Reactor Regulation (NRR) responsible for the primary and secondary reviews, and (3) the applicable guidance documents to be used for performing the reviews. Acceptance criteria for the reviews are included in the referenced guidance documents.

The review of an EPU application involves the following three steps:

Step 1. Initial Screening

Upon receipt of an EPU application, the Project Manager will conduct an initial screening of the application for completeness and acceptability consistent with the guidance in NRR Office Instruction LIC-101, "License Amendment Review Procedures." This review is conducted to ensure that the application meets the minimum requirements described in 10 CFR 50.4, 10 CFR 50.90, 10 CFR 50.91, and 10 CFR 50.92. The Project Manager will distribute the application to the technical staff and proceed with the acceptance review if the application meets the minimum requirements.

Step 2. Acceptance Review

The Project Manager will review the EPU application to ensure that it adequately identifies the licensing basis of the plant for the items in the "Areas of Review" column in the matrices. The Project Manager should coordinate this effort with the acceptance review conducted by the reviewers with the primary review responsibility (discussed below).

Reviewers with primary review responsibility should follow the instructions below for completing the acceptance review.

(1) Based on the information provided in the EPU application, annotate the items in the "Areas of Review" column in the matrices to indicate (a) applicability of the items to the plant under review, (b) any additional areas of review that are affected by the EPU (as identified in the EPU application), and (c) any beyond-scope items that are included in the EPU application. (Licensees are encouraged to complete the matrices as part of their application as a quality check to assure that all necessary information has been provided and properly represented, thereby avoiding potential delays and improving the efficiency of the staff's review.)

(2) Conduct an acceptance review to confirm that the licensee has addressed the applicable areas identified in the "Areas of Review" column of the matrices (as modified based on instruction (1) above). Review the information provided by the licensee for each area of review that is affected by the EPU to confirm that the regulatory requirements and licensing basis are adequately characterized and addressed with respect to the proposed EPU.

(3) Use the "Acceptance Review" column of the matrices as a checklist to document whether the licensee has addressed the areas of review in sufficient detail to allow the staff to proceed with its detailed technical review. Any negative comments in this column may lead to the NRC staff's denial of the application, or in substantial schedule delays.

(4) Before proceeding with the detailed technical review, provide the plant Project Manager a copy of the matrix completed as a result of instruction (3) above.

Step 3. Detailed Technical Review

(1) Compare the guidance in the documents referenced in the "SRP Section Number" and "Other Guidance" columns of the matrices to the licensing basis of the plant as described in the EPU application for each item in the "Areas of Review" column. Use the "Focus of SRP Usage" column to identify the applicable portions of the SRP sections identified. If the licensing basis of the plant that is identified in the EPU application is different from the guidance provided in the documents referenced in the matrices, consult with the Project Manager regarding the differences and compliance of the information in the EPU application with applicable regulations. Revise the matrices, as appropriate, based on the results of the review.

(2) If the areas of review for the plant are determined to be different from the areas identified in the matrices, obtain oral concurrence from the branch chief of the primary review branch for the differences. This should be done for additions to as well as deletions from the list of items in the "Areas of Review" column.

(3) Provide the revised matrices to the Project Manager. (Licensees are encouraged to complete the matrices as part of their application as a quality check to assure that all necessary information has been provided and properly represented, thereby avoiding potential delays and improving the efficiency of the staff's review.)

(4) Conduct a detailed review of the application consistent with the guidance provided in the documents listed in the "SRP Section Number" and "Other Guidance" columns (as modified to suit the licensing basis of the plant). Use the "Focus of SRP Usage" column to identify the applicable portions of the SRP sections identified.

(5) Coordinate with the technical branches identified in the "Secondary Review Branch(es)" column to ensure that all important aspects of each technical area are adequately covered during the review.

(6) Perform independent calculations consistent with the guidance in Attachment 1 to each matrix. Any issues identified by the NRC staff as a result of its independent calculations should be resolved with the licensee. If necessary, the licensee should be requested to update and resubmit any affected analyses. It should be noted that the NRC staff's approval of the application is to be based on the licensee's docketed information.

(7) Document the results of the detailed technical review in accordance with the guidance in Section 3.1 of this review standard.

MATRIX 1

SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

Materials and Chemical Engineering

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review Checklist
							BWR	PWR	
Reactor Vessel Material Surveillance Program	All EPUs	EMCB	SRXB	5.3.1 Draft Rev. 2 April 1996	GDC-14 GDC-31 10 CFR 50, App. H 10 CFR 50.60	RG 1.190	2.1.1	2.1.1	
Pressure-Temperature Limits and Upper-Shelf Energy	All EPUs	EMCB	SRXB	5.3.2 Draft Rev. 2 April 1996	GDC-14 GDC-31 10 CFR 50, App. G 10 CFR 50.60	RG 1.161 RG 1.190 RG 1.99	2.1.2	2.1.2	
Pressurized Thermal Shock	PWR EPUs	EMCB	SRXB	5.3.2 Draft Rev. 2 April 1996	GDC-14 GDC-31 10 CFR 50.61	RG 1.190 RG 1.154		2.1.3	
Reactor Internal and Core Support Materials	All EPUs	EMCB	SRXB	4.5.2 Draft Rev. 3 April 1996	GDC-1 10 CFR 50.55a	Note 1*	2.1.3	2.1.4	

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review Checklist
							BWR	PWR	
Reactor Coolant Pressure Boundary Materials	All EPUs	EMCB	EMEB SRXB	5.2.3 Draft Rev. 3 April 1996	GDC-1 10 CFR 50.55a GDC-4 GDC-14 GDC-31 10 CFR 50, App. G	RG 1.190 GL 97-01 IN 00-17s1 BL 01-01 BL 02-01 BL 02-02 Note 2* Note 3*	2.1.4	2.1.5	
				4.5.1 Draft Rev. 3 April 1996	GDC-1 10 CFR 50.55a GDC-14				
				5.2.4 Draft Rev. 2 April 1996	10 CFR 50.55a				
				5.3.1 Draft Rev. 2 April 1996	GDC-1 10 CFR 50.55a GDC-4 GDC-14 GDC-31 10 CFR 50, App. G				
				5.3.3 Draft Rev. 2 April 1996					
				6.1.1 Draft Rev. 2 April 1996					
Leak-Before-Break	PWR EPUs	EMCB		3.6.3 Draft Aug. 1987	GDC-4	NUREG 1061 Vol. 3 Nov. 1984		2.1.6	
Protective Coating Systems (Paints) - Organic Materials	All EPUs	EMCB		6.1.2 Draft Rev. 3 April 1996	10 CFR 50, App. B RG 1.54		2.1.5	2.1.7	

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review Checklist
							BWR	PWR	
Effect of EPU on Flow-Accelerated Corrosion	All EPUs	EMCB				Note 4*	2.1.6	2.1.8	
Steam Generator Tube Inservice Inspection	PWR EPUs	EMCB		5.4.2.2 Draft Rev. 2 April 1996	10 CFR 50.55a	Plant TSs RG 1.121 GL 95-03 BL 88-02 GL 95-05 Note 5*		2.1.9	
Steam Generator Blowdown System	PWR EPUs	EMCB		10.4.8 Draft Rev. 3 April 1996	GDC-14			2.1.10	
Chemical and Volume Control System (Including Boron Recovery System)	PWR EPUs	EMCB	SPLB SRXB	9.3.4 Draft Rev. 3 April 1996	GDC-14 GDC-29			2.1.11	
Reactor Water Cleanup System	BWR EPUs	EMCB		5.4.8 Draft Rev. 3 April 1996	GDC-14 GDC-60 GDC-61		2.1.7		

Notes:

1. In addition to the SRP, guidance on the neutron irradiation-related threshold for inspection for irradiation-assisted stress-corrosion cracking for BWRs is in BWRVIP-26 and for PWRs in BAW-2248 for E>1 MeV and in WCAP-14577 for E>0.1 MeV. For intergranular stress-corrosion cracking and stress-corrosion cracking in BWRs, review criteria and review guidance is contained in BWRVIP reports and associated staff safety evaluations. For thermal and neutron embrittlement of cast austenitic stainless steel, stress-corrosion cracking, and void swelling, licensees will need to provide plant-specific degradation management programs or participate in industry programs to investigate degradation effects and determine appropriate management programs.

2. For thermal aging of cast austenitic stainless steel, review guidance and criteria is contained in the May 19, 2000, letter from C. Grimes to D. Walters, "Thermal Aging Embrittlement of Cast Austenitic Stainless Steel Components."

3. For intergranular stress corrosion cracking in BWR piping, review criteria and review guidance is contained in BWRVIP reports, NUREG-0313, Revision 2, GL 88-01, Supplement 1 to GL-88-01, and associated safety evaluations.

4. Criteria and review guidance needed to review EPU applications in the area of flow-accelerated corrosion is contained in Electric Power Research Institute (EPRI) Report NSAC-202L-R2, "Recommendations for Effective an Flow-Accelerated Corrosion Program," dated April 1999. This EPRI document is copyrighted. EPRI has provided copies of this document to EMCB for use by NRC staff. Copying of this document, however, is not allowed.

5. Also see the plant-specific license amendments approving alternate repair criteria and redefining inspection boundaries.

LIST OF ACRONYMS FOR MATRIX 1

BL = bulletin
BWR = boiling-water reactor
CFR = *Code of Federal Regulations*
EMCB = Materials & Chemical Engineering Branch
EMEB = Mechanical & Civil Engineering Branch
EPUs = extended power uprates
GDC = General Design Criterion
GL = generic letter
PWR = pressurized-water reactor
RG = regulatory guide
SPLB = Plant Systems Branch
SRP = Standard Review Plan
SRXB = Reactor Systems Branch

ATTACHMENT 1 TO MATRIX 1

Independent Calculations

Materials and Chemical Engineering

Perform independent calculations of the pressurized thermal shock reference temperature and upper-shelf energy (if there is a change in the evaluation of these quantities as a result of the proposed extended power uprate).

MATRIX 2

SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

Mechanical and Civil Engineering

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review Checklist
							BWR	PWR	
Pipe Rupture Locations and Associated Dynamic Effects	All EPUs	EMEB		3.6.2 Draft Rev. 2 April 1996	GDC-4		2.2.1	2.2.1	

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review Checklist
							BWR	PWR	
Pressure-Retaining Components and Component Supports	All EPUs	EMEB		3.9.1 Draft Rev. 3 April 1996	GDC-1 GDC-2 GDC-14 GDC-15		2.2.2	2.2.2	
				3.9.2 Draft Rev. 3 April 1996	GDC-1 GDC-2 GDC-4 GDC-14 GDC-15	IN 95-016 IN 02-026			
				3.9.3 Draft Rev. 2 April 1996	10 CFR 50.55a GDC-1 GDC-2 GDC-4 GDC-14 GDC-15	IN 96-049 GL 96-06			
				5.2.1.1 Draft Rev. 3 April 1996	10 CFR 50.55a GDC-1	RG 1.84 RG 1.147 DG 1.1089 DG 1.1090 DG 1091			

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review Checklist
							BWR	PWR	
Reactor Pressure Vessel Internals and Core Supports	All EPUs	EMEB		3.9.1 Draft Rev. 3 April 1996	GDC-1 GDC-2		2.2.3	2.2.3	
				3.9.2 Draft Rev. 3 April 1996	GDC-1 GDC-2 GDC-4	IN 95-016 IN 02-026			
				3.9.3 Draft Rev. 2 April 1996	10 CFR 50.55a GDC-1 GDC-2 GDC-4	IN 96-049 GL 96-06			
				3.9.5 Draft Rev. 3 April 1996	10 CFR 50.55a GDC-1 GDC-2 GDC-4 GDC-10	IN 02-026 Note 1*			

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review Checklist
							BWR	PWR	
Safety-Related Valves and Pumps	All EPUs	EMEB		3.9.3 Draft Rev. 2 April 1996	GDC-1 10 CFR 50.55a(f)	IN 96-049 GL 96-06	2.2.4	2.2.4	
				3.9.6 Draft Rev. 3 April 1996	GDC-1 GDC-37 GDC-40 GDC-43 GDC-46 GDC-54 10 CFR 50.55a(f)	GL 89-10 GL 95-07 GL 96-05 IN 97-090 IN 96-048s1 IN 96-048 IN 96-003 RIS 00-003 RIS 01-015 RG 1.147 RG 1.175 DG 1089 DG 1091			
Seismic and Dynamic Qualification of Mechanical and Electrical Equipment	All EPUs	EMEB	EEIB	3.10 Draft Rev. 3 April 1996	GDC-1 GDC-2 GDC-4 GDC-14 GDC-30 10 CFR 100, App. A 10 CFR 50, App. B USI A-46		2.2.5	2.2.5	

Notes:

1. In addition to the failure of steam dryer lower cover plate as reported in NRC IN 02-26, recent steam dryer failure with cracks on the outer dryer hood and top cover plate at Quad Cities Unit 2 was identified due to low frequency flow induced vibration loading. The staff's review of the reactor internals will cover detailed analysis of flow induced vibration and acoustic vibration (where applicable) on reactor internal components such as steam dryers and separators, and the jet pump sensing lines that are affected by the increased steam and feedwater flow for the extended power uprate.

LIST OF ACRONYMS FOR MATRIX 2

BWR = boiling-water reactor
CFR = *Code of Federal Regulations*
DG = draft guide
EEIB = Electrical & Instrumentation & Controls Branch
EMEB = Mechanical & Civil Engineering Branch
EPUs = extended power uprates
GDC = General Design Criterion
GL = generic letter
IN = information notice
PWR = pressurized-water reactor
RG = regulatory guide
RIS = regulatory issue summary
SRP = Standard Review Plan

ATTACHMENT 1 TO MATRIX 2

Independent Calculations

Mechanical and Civil Engineering

Independent calculations are not performed in the area of mechanical engineering. However, audits of the licensee's calculations should be performed, as necessary, to verify that the licensee's application of the methodologies is correct and consistent with NRC staff positions.

MATRIX 3

SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

Electrical Engineering

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review Checklist
							BWR	PWR	
Environmental Qualification of Electrical Equipment	All EPUs	EEIB		3.11 Draft Rev. 3 April 1996	10 CFR 50.49		2.3.1	2.3.1	
Offsite Power System	All EPUs	EEIB		8.1 Draft Rev. 3 April 1996	GDC-17	BTP PSB-1 Draft Rev. 3 April 1996 BTP ICSB-11 Draft Rev. 3 April 1996	2.3.2	2.3.2	
				8.2 Draft Rev. 4 April 1996	GDC-17				
				8.2, App. A Draft Rev. 4 April 1996	GDC-17				
AC Onsite Power System	All EPUs	EEIB		8.1 Draft Rev. 3 April 1996	GDC-17		2.3.3	2.3.3	
				8.3.1 Draft Rev. 3 April 1996	GDC-17				

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review Checklist
							BWR	PWR	
DC Onsite Power System	All EPUs	EEIB		8.1 Draft Rev. 3 April 1996	GDC-17 10 CFR 50.63		2.3.4	2.3.4	
				8.3.2 Draft Rev. 3 April 1996	GDC-17 10 CFR 50.63				
Station Blackout	All EPUs	EEIB	SPLB SRXB	8.1 Draft Rev. 3 April 1996	10 CFR 50.63	Note 1*	2.3.5	2.3.5	
				8.2, App. B Draft Rev. 4 April 1996	10 CFR 50.63				

1. The review of station blackout includes the effects of the EPU on systems required for core cooling in the station blackout coping analysis (e.g., condensate storage tank inventory, controls and power supplies for relief valves, residual heat removing system) to ensure that the effects are accounted for in the analysis.

LIST OF ACRONYMS FOR MATRIX 3

BWR = boiling-water reactor
CFR = *Code of Federal Regulations*
EEIB = Electrical & Instrumentation & Controls Branch
EPUs = extended power uprates
GDC = General Design Criterion
PWR = pressurized-water reactor
SRP = Standard Review Plan
BTP = branch technical position
AC = alternating current
DC = direct current

ATTACHMENT 1 TO MATRIX 3

Independent Calculations

Electrical Engineering

Independent calculations are not performed in the area of electrical engineering. However, the following should be verified to ensure that reliable power sources continue to be available to safety buses following implementation of the proposed extended power uprate:

- capability curve of the main generator
- grid stability contingencies
- capability of the isophase bus and the transformers

MATRIX 4

SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

Instrumentation and Controls

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review Checklist
							BWR	PWR	
Reactor Trip System	All EPUs	EEIB		7.2 Rev. 4 June 1997	10 CFR 50.55(a)(1) 10 CFR 50.55a(h) GDC-1 GDC-4 GDC-13 GDC-19 GDC-20 GDC-21 GDC-22 GDC-23 GDC-24		2.4.1	2.4.1	
Engineered Safety Features Systems	All EPUs	EEIB		7.3 Rev. 4 June 1997			2.4.1	2.4.1	
Safety Shutdown Systems	All EPUs	EEIB		7.4 Rev. 4 June 1997	10 CFR 50.55(a)(1) 10 CFR 50.55a(h) GDC-1 GDC-4 GDC-13 GDC-19 GDC-24		2.4.1	2.4.1	

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review Checklist
							BWR	PWR	
Control Systems	All EPUs	EEIB		7.7 Rev. 4 June 1997	10 CFR 50.55(a)(1) 10 CFR 50.55a(h) GDC-1 GDC-13 GDC-19 GDC-24		2.4.1	2.4.1	
Diverse I&C Systems	All EPUs	EEIB		7.8 Rev. 4 June 1997			2.4.1	2.4.1	
General guidance for use of other SRP Sections related to I&C	All EPUs	EEIB		7.0 Rev. 4 June 1997					

LIST OF ACRONYMS FOR MATRIX 4

BWR = boiling-water reactor
CFR = *Code of Federal Regulations*
EEIB = Electrical & Instrumentation & Controls Branch
EPUs = extended power uprates
GDC = General Design Criterion
I&C = instrumentation and controls
PWR = pressurized-water reactor
SRP = Standard Review Plan

ATTACHMENT 1 TO MATRIX 4

Independent Calculations

Instrumentation and Controls

Independent calculations are not performed in the area of instrumentation and controls. For a plant where an instrument setpoint methodology has not been previously approved, a detailed review of the licensee's calculations for one instrument should be performed to verify that the licensee's application of the methodologies is correct and consistent with NRC staff positions.

MATRIX 5

SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

Plant Systems

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review Checklist
							BWR	PWR	
Flood Protection	EPUs that result in significant increases in fluid volumes of tanks and vessels	SPLB		3.4.1 Rev. 2 July 1981	GDC-2		2.5.1.1.1	2.5.1.1.1	
Equipment and Floor Drainage System	EPUs that result in increases in fluid volumes or in installation of larger capacity pumps or piping systems	SPLB		9.3.3 Rev. 2 July 1981	GDC-2 GDC-4		2.5.1.1.2	2.5.1.1.2	
Circulating Water System	EPUs that result in increases in fluid volumes associated with the circulating water system or in installation of larger capacity pumps or piping systems	SPLB		10.4.5 Rev. 2 July 1981	GDC-4		2.5.1.1.3	2.5.1.1.3	
Internally Generated Missiles (Outside Containment)	EPUs that result in substantially higher system pressures or changes in existing system configuration	SPLB	EMCB EMEB	3.5.1.1 Rev. 2 July 1981	GDC-4		2.5.1.2.1	2.5.1.2.1	
Internally Generated Missiles (Inside Containment)	EPUs that result in substantially higher system pressures or changes in existing system configuration	SPLB	EMCB EMEB	3.5.1.2 Rev. 2 July 1981	GDC-4		2.5.1.2.1	2.5.1.2.1	

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review Checklist
							BWR	PWR	
Turbine Generator	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB		10.2 Rev. 2 July 1981	GDC-4		2.5.1.2.2	2.5.1.2.2	
Protection Against Postulated Piping Failures in Fluid Systems Outside Containment	EPUs that affect environmental conditions, habitability of the control room, or access to areas important to safe control of postaccident operations	SPLB	EMCB EMEB	3.6.1 Rev. 1 July 1981	GDC-4		2.5.1.3	2.5.1.3	
Fire Protection Program	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB		9.5.1 Rev. 3 July 1981	10 CFR 50.48 10 CFR 50, App. R GDC-3 GDC-5	Note 1*	2.5.1.4	2.5.1.4	
Pressurizer Relief Tank	PWR EPUs that affect pressurizer discharge to the PRT	SPLB	EMEB	5.4.11 Rev. 2 July 1981	GDC-2 GDC-4			2.5.2	
Fission Product Control Systems and Structures	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB	EMCB	6.5.3 Rev. 2 July 1981	GDC-41		2.5.2.1	2.5.3.1	
Main Condenser Evacuation System	EPUs for which the main condenser evacuation system is modified	SPLB		10.4.2 Rev. 2 July 1981	GDC-60 GDC-64		2.5.2.2	2.5.3.2	
Turbine Gland Sealing System	EPUs for which the turbine gland sealing system is modified	SPLB		10.4.3 Rev. 2 July 1981	GDC-60 GDC-64		2.5.2.3	2.5.3.3	
Main Steam Isolation Valve Leakage Control System	BWR EPU that affect the amount of valve leakage that is assumed and resultant dose consequences.	SPLB		6.7 Rev. 2 July 1981	GDC-54		2.5.2.4		

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review Checklist
							BWR	PWR	
Spent Fuel Pool Cooling and Cleanup System	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB	EMCB	9.1.3 Rev. 1 July 1981	GDC-5 GDC-44 GDC-61	Note 2*	2.5.3.1	2.5.4.1	
Station Service Water System	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB		9.2.1 Rev. 4 June 1985	GDC-4 GDC-5 GDC-44	GL 89-13 and Suppl. 1 GL 96-06 and Suppl. 1	2.5.3.2	2.5.4.2	
Reactor Auxiliary Cooling Water Systems	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB		9.2.2 Rev. 3 June 1986	GDC-4 GDC-5 GDC-44	GL 89-13 and Suppl. 1 GL 96-06 and Suppl. 1	2.5.3.3	2.5.4.3	
Ultimate Heat Sink	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB		9.2.5 Rev. 2 July 1981	GDC-5 GDC-44		2.5.3.4	2.5.4.4	
Auxiliary Feedwater System	PWR EPUs except where the application demonstrates that previous analysis is bounding	SPLB		10.4.9 Rev. 2 July 1981	GDC-4 GDC-5 GDC-19 GDC-34 GDC-44			2.5.4.5	
Main Steam Supply System	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB		10.3 Rev. 3 April 1984	GDC-4 GDC-5 GDC-34		2.5.4.1	2.5.5.1	

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review Checklist
							BWR	PWR	
Main Condenser	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB		10.4.1 Rev. 2 July 1981	GDC-60		2.5.4.2	2.5.5.2	
Turbine Bypass System	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB		10.4.4 Rev. 2 July 1981	GDC-4 GDC-34		2.5.4.3	2.5.5.3	
Condensate and Feedwater System	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB		10.4.7 Rev. 3 April 1984	GDC-4 GDC-5 GDC-44		2.5.4.4	2.5.5.4	
Gaseous Waste Management Systems	EPUs that impact the level of fission products in the reactor coolant system, or the amount of gaseous waste	SPLB	IEPB	11.3 Draft Rev. 3 April 1996	10 CFR 20.1302 GDC-3 GDC-60 GDC-61 10 CFR 50, App. I		2.5.5.1	2.5.6.1	
Liquid Waste Management Systems	EPUs that impact the level of fission products in the reactor coolant system, or the amount of liquid waste	SPLB	IEPB	11.2 Draft Rev. 3 April 1996	10 CFR 20.1302 GDC-60 GDC-61 10 CFR 50, App. I		2.5.5.2	2.5.6.2	
Solid Waste Management Systems	EPUs that impact the level of fission products in the reactor coolant system, or the amount of solid waste	SPLB	IEPB	11.4 Draft Rev. 3 April 1996	10 CFR 20.1302 GDC-60 GDC-63 GDC-64 10 CFR 71		2.5.5.3	2.5.6.3	
Emergency Diesel Engine Fuel Oil Storage and Transfer System	EPUs that result in higher EDG electrical demands	SPLB		9.5.4 Rev. 2 July 1981	GDC-4 GDC-5 GDC-17		2.5.6.1	2.5.7.1	
Light Load Handling System (Related to Refueling)	EPUs except where the application demonstrates that previous analysis is bounding	SPLB	SPSB	9.1.4 Rev. 2 July 1981	GDC-61 GDC-62		2.5.6.2	2.5.7.2	

Notes:

1. Supplemental guidance for review of fire protection is provided in Attachment 2 to this matrix.

2. Supplemental guidance for review of spent fuel pool cooling is provided in Attachment 3 to this matrix.

LIST OF ACRONYMS FOR MATRIX 5

BWR = boiling-water reactor
CFR = *Code of Federal Regulations*
EMCB = Materials & Chemical Engineering Branch
EMEB = Mechanical & Civil Engineering Branch
EPUs = extended power uprates
GDC = General Design Criterion
GL = generic letter
IEPB = Emergency Preparedness and Plant Support Branch
PWR = pressurized-water reactor
SPLB = Plant Systems Branch
SPSB = Probabalistic Safety Assessment Branch
SRP = Standard Review Plan

ATTACHMENT 1 TO MATRIX 5

Independent Calculations

Plant Systems

Use the criteria in the Standard Review Plan sections referenced in Matrix 5 for determining when to perform independent calculations.

ATTACHMENT 2 TO MATRIX 5

Supplemental Fire Protection Review Criteria

Plant Systems

This attachment provides guidance for the review of the fire protection information to be provided in an application for a power uprate. Power uprates typically result in increases in decay heat generation following plant trips. These increases in decay heat usually do not affect the elements of a fire protection program related to (1) administrative controls, (2) fire suppression and detection systems, (3) fire barriers, (4) fire protection responsibilities of plant personnel, and (5) procedures and resources necessary for the repair of systems required to achieve and maintain cold shutdown. In addition, an increase in decay heat will usually not result in an increase in the potential for a radiological release resulting from a fire. However, the licensee's application should confirm that these elements are not impacted by the extended power uprate. This confirmation should be reflected in the staff's safety evaluation. If the licensee indicates that there is an impact on these elements, the staff should review the licensee's assessment of the impact using this attachment.

The systems relied upon to achieve and maintain safe shutdown following a fire may be affected by the power uprate due to the increase in decay heat generation following a plant trip. For fire events where the licensee is relying on one full train of the redundant systems normally used for safe shutdown, the analysis of the impact of the power uprate on the important plant process parameters performed for other plant transients (such as a loss of offsite power or a loss of main feedwater) will typically bound the impact of a fire event. In this case, a specific analysis for fire events may not be required. However, where licensees rely on less than full capability systems for fire events (e.g., partial automatic depressurization system capability for reduced capability makeup pump), the licensee should provide specific analyses for fire events that demonstrate that (1) fuel integrity is maintained by demonstrating that the fuel design limits are not exceeded and (2) there are no adverse consequences on the reactor pressure vessel integrity or the attached piping. Plants that rely on alternative/dedicated or backup shutdown capability for post-fire safe shutdown should analyze the impact of the power uprate on the alternative/dedicated or backup shutdown capability. The staff should verify that the capability of the alternative/dedicated or backup systems relied upon for post-fire safe shutdown are capable of achieving and maintaining safe shutdown considering the impact of the power uprate.

The plant's post-fire safe shutdown procedures may also be impacted by the power uprate. For example, the allowable time to perform necessary operator actions may decrease as a result of the power uprate. In this case, the required flow rates for systems required to achieve and maintain safe shutdown may need to be increased. The licensee should identify the impact of the power uprate on the plant's post-fire safe shutdown procedures.

ATTACHMENT 3 TO MATRIX 5

Supplemental Spent Fuel Pool Cooling Review Criteria

Plant Systems

1. BACKGROUND

All operating nuclear power plants were licensed to certain design criteria regarding the adequacy of spent fuel pool (SFP) cooling capability. The most common criterion is that contained in General Design Criterion (GDC)-61 of Appendix A to 10 CFR Part 50. This criterion specifies, in part, that the fuel storage system (1) be designed with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal and (2) be designed to prevent a significant reduction in coolant inventory under accident conditions. Earlier licensing criteria are consistent with the intent of GDC-61. However, later guidance contained in Section 9.1.3 of the Standard Review Plan invoked GDC-44 for the SFP cooling system, which specifies provision of a redundant cooling system that is capable of operation with or without offsite sources of power. To satisfy these criteria, each licensee must demonstrate that there is adequate SFP cooling capacity and the ability to supply adequate make-up water in the event of total loss of SFP cooling.

A significant design-basis challenge to the SFP cooling system is imposed by a planned evolution (fuel transfer from the reactor vessel). Emergency offloads are not considered credible because fuel transfers require plant cooldown, reactor disassembly, and refueling cavity flooding, which are time-consuming, manual processes. As a result, the staff will review factors that increase heat load (e.g., power increases, decay-time reductions, or storage capacity increases) and other operational factors that reduce heat load (e.g., longer decay times or transfer of fewer fuel assemblies to the SFP) or that increase heat removal capability (e.g., scheduling offloads for periods of reduced ultimate heat sink temperature or optimizing cooling system performance).

This guidance supercedes the guidance of paragraphs III.1.d. and III.1.h. of Standard Review Plan Section 9.1.3.

2. ACCEPTANCE CRITERIA

The adequacy of cooling may be evaluated against the capability to complete normal, planned activities, including fuel handling, without a degradation in safety and the ability to maintain defense-in-depth against a significant reduction in coolant inventory under accident conditions. With respect to fuel handling, which is a manual process, SFP temperatures affect safety through operating environment and visibility. At SFP temperatures below 140°F, (1) the fuel handling building ventilation is typically adequate to maintain a suitable operating environment, (2) evaporation from the SFP surface is at a sufficiently low rate to preclude fogging, and (3) the SFP temperature is within the design range of the cleanup system demineralizes to maintain water clarity. Defense-in-depth is provided by:

(1) alarms to notify operators of a loss of cooling;
(2) the capability of the SFP cooling system to maintain or reestablish, within a reasonable time, forced cooling following a single failure of an active component;
(3) the ability of the cooling system to maintain the SFP temperature below the design temperature of the SFP structure and liner following a single-active failure or a design-basis event (e.g., a seismic event) within the current licensing basis of the facility; and
(4) the availability of two reliable sources of makeup water, one of which having the capacity to make up for evaporation following a total loss of forced cooling.

The reliability of the systems relied upon to meet these guidelines should be maintained consistent with the plant's current licensing basis.

3. REVIEW PROCEDURES

3.1. Adequate SFP Cooling Capacity

The licensee demonstrates adequate SFP cooling capacity by either performing a bounding evaluation or committing to a method of performing outage-specific evaluations.

3.1.1. Bounding Calculation

Two scenarios are analyzed: (1) full cooling capability and (2) a single failure of an active cooling system component.

3.1.1.1. Full Cooling System Capability Evaluation

Analysis conditions:

(1) decay heat load is calculated based on bounding estimates of offload size, decay time, power history, and inventory of previously discharged assemblies
(2) heat removal capability is based on bounding estimates of ultimate heat sink temperature, cooling system flow rates, and heat exchanger performance (e.g., fouling and tube plugging margin)
(3) alternate heat removal paths (e.g., evaporative cooling) must be appropriately validated and based on bounding input parameter values (e.g., air temperature, relative humidity, and ventilation flow rate)
(4) actual bulk SFP temperature must remain below 140 °F - calculated SFP temperatures up to approximately 150 °F are acceptable when justified by conservative methods or assumptions
(5) with appropriate administrative controls to verify that analysis inputs bound actual conditions, a set of bounding analyses may be prepared by the licensee to support operational flexibility.

3.1.1.2. Single-Active Failure Evaluation

Analysis conditions:

(1) decay heat load is calculated based on a bounding estimate of offload size, decay time, power history, and inventory of previously discharged assemblies
(2) heat removal capability is based on a bounding estimate of ultimate heat sink temperature, heat exchanger performance (e.g., fouling and tube plugging margin), and cooling system flow rates assuming the limiting single failure with regard to heat removal capability
(3) alternate heat removal paths (e.g., evaporative cooling) must be appropriately validated and based on bounding input parameter values (e.g., air temperature, relative humidity, and ventilation flow rate)
(4) calculated bulk SFP temperature must remain below the design temperature of the SFP structure and liner, and calculated peak storage cell temperature must remain below the storage rack design temperature
(5) for plants where a single failure results in a complete loss of forced cooling, the licensee's analysis should demonstrate that the loss of cooling would be identified and forced cooling would be restored before the bounding decay heat load would cause the SFP temperature to reach its design limit
(6) with appropriate administrative controls to verify that analysis inputs bound actual conditions, a set of bounding analyses may be prepared by the licensee to support operational flexibility.

3.1.2. Cycle-Specific Calculation:

The licensee can choose to define a method to calculate operational limits prior to every offload using the anticipated actual conditions at the time of the offload.

Cycle-specific analysis conditions:

(1) define the method to calculate decay heat load based on decay time, power history, and inventory of previous fuel discharges
(2) define the method to calculate cooling system heat removal capacity based on ultimate heat sink temperature, cooling system flow rates, and heat exchanger performance parameters
(3) define the method for calculating alternate heat removal capability (e.g., evaporative cooling) and provide validation of the method
(4) using the methods defined to calculate heat load and heat removal capability, define the method to determine the limiting value of the variable operational parameter (typically, decay time) such that bulk SFP temperature will remain below 140 °F with full cooling capability
(5) using the methods defined to calculate heat load and heat removal capability, define the method to determine the limiting value of the variable operational parameter (typically, decay time) such that bulk SFP temperature will be maintained below the SFP structure design temperature assuming a single failure affecting the forced cooling system (this may be a heat-balance analysis if cooling is degraded or a heatup-rate analysis if forced cooling is completely lost and subsequently recovered using redundant components)

(6) describe administrative controls that will be implemented each offload to ensure the cycle-specific analysis inputs and results bound actual conditions prior to fuel movement

3.2. Adequate Make-Up Supply

(1) Following a loss-of-SFP cooling event, the licensee must be able to provide two sources of make-up water prior to the occurrence of boiling in the pool. To determine the time to boil, the initial pool temperature is the peak temperature from a planned offload, assuming the worst single-active failure occurred.

(2) At least one make-up source shall have a capacity that is equal to or greater than the calculated boil-off rate so that the SFP level can be maintained.

MATRIX 6

SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

Containment Review Considerations

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review Checklist
							BWR	PWR	
PWR Dry Containments, Including Subatmospheric Containments	EPUs for PWR plants with dry containments (including subatmospheric containments) except where the application demonstrates that previous analysis is bounding	SPSB		6.2.1 Rev. 2 July 1981 6.2.1.1.A Rev. 2 July 1981	GDC-13 GDC-16 GDC-38 GDC-50 GDC-64			2.6.1	
Ice Condenser Containments	EPUs for PWR plants with ice condenser containments except where the application demonstrates that previous analysis is bounding	SPSB		6.2.1 Rev. 2 July 1981 6.2.1.1.B Rev. 2 July 1981	GDC-13 GDC-16 GDC-38 GDC-50 GDC-64			2.6.1	

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review Checklist
							BWR	PWR	
Pressure-Suppression Type BWR Containments	EPUs for BWR plants with pressure-suppression containments except where the application demonstrates that previous analysis is bounding	SPSB		6.2.1 Rev. 2 July 1981 6.2.1.1.C Rev. 6 Aug. 1984	GDC-4 GDC-13 GDC-16 GDC-50 GDC-64		2.6.1		
Subcompartment Analysis	All EPUs except where the application demonstrates that previous analysis is bounding	SPSB		6.2.1 Rev. 2 July 1981 6.2.1.2 Rev. 2 July 1981	GDC-4 GDC-50		2.6.2	2.6.2	
Mass and Energy Release Analysis for Postulated Loss-of-Coolant	All EPUs except where the application demonstrates that previous analysis is bounding	SPSB		6.2.1 Rev. 2 July 1981 6.2.1.3 Rev. 1 July 1981	GDC-50 10 CFR 50, App. K		2.6.3.1	2.6.3.1	
Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures	PWR EPUs except where the application demonstrates that previous analysis is bounding	SPSB		6.2.1 Rev. 2 July 1981 6.2.1.4 Rev. 1 July 1981	GDC-50			2.6.3.2	

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review Checklist
							BWR	PWR	
Combustible Gas Control In Containment	EPUs that impact hydrogen release assumptions	SPSB		6.2.5 Rev. 2 July 1981	10 CFR 50.44 10 CFR 50.46 GDC-5 GDC-41 GDC-42 GDC-43		2.6.4	2.6.4	
Containment Heat Removal	All EPUs except where the application demonstrates that previous analysis is bounding	SPSB		6.2.2 Rev. 4 Oct. 1985	GDC-38	DG-1107	2.6.5	2.6.5	
Secondary Containment Functional Design	EPUs that affect the pressure and temperature response, or draw-down time of the secondary containment	SPSB		6.2.3 Rev. 2 July 1981	GDC-4 GDC-16		2.6.6		
Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies	PWR EPUs except where the application demonstrates that previous analysis is bounding	SPSB	SRXB	6.2.1 Rev. 2 July 1981 6.2.1.5 Rev. 2 July 1981	10 CFR 50.46 10 CFR 50, App. K			2.6.6	

LIST OF ACRONYMS FOR MATRIX 6

BWR = boiling-water reactor
CFR = *Code of Federal Regulations*
DG = draft guide
EPUs = extended power uprates
GDC = General Design Criterion
PWR = pressurized-water reactor
SPSB = Probabalistic Safety Assessment Branch
SRP = Standard Review Plan
SRXB = Reactor Systems Branch

ATTACHMENT 1 TO MATRIX 6

Independent Calculations

Containment Review Considerations

Use the following guidelines for determining when to perform independent calculations:

- The licensee has performed analyses which have changed substantially since they were approved or used in plants at similar power levels
- The licensee has performed analyses using methods which have not been previously used at the plant or at a similar plant at similar power levels
- The licensee has performed a type of analysis (e.g., subcompartment pressure-temperature, water level, EQ envelope) that has not been previously reviewed by the staff for that application
- The licensee has performed analyses using first-of-a-kind methods
- The licensee has not adequately addressed the impact of the proposed extended power uprate on the assumptions, range of applicability, or suitability of the methods used for the analyses
- The results of the licensee's analysis are questionable in light of (1) NRC staff review experience, (2) the results of other NRC staff calculations, or (3) the results of current or previous research activities
- The licensee's analyses show significant reductions in available margin

MATRIX 7

SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

Habitability, Filtration, and Ventilation

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review Checklist
							BWR	PWR	
Control Room Habitability System	All EPUs except where the application demonstrates that previous analysis is bounding	SPSB		6.4 Draft Rev. 3 April 1996	GDC-4 GDC-19	Note 1* Note 2*	2.7.1	2.7.1	
ESF Atmosphere Cleanup System	All EPUs except where the application demonstrates that previous analysis is bounding	SPSB		6.5.1 Rev. 2 July 1981	GDC-19 GDC-41 GDC-61 GDC-64		2.7.2	2.7.2	
Control Room Area Ventilation System	All EPUs except where the application demonstrates that previous analysis is bounding	SPSB		9.4.1 Rev. 2 July 1981	GDC-4 GDC-19 GDC-60		2.7.3	2.7.3	
Spent Fuel Pool Area Ventilation System	All EPUs except where the application demonstrates that previous analysis is bounding	SPSB		9.4.2 Rev. 2 July 1981	GDC-60 GDC-61		2.7.4	2.7.4	
Auxiliary and Radwaste Area Ventilation System	All EPUs except where the application demonstrates that previous analysis is bounding	SPSB		9.4.3 Rev. 2 July 1981	GDC-60		2.7.5	2.7.5	
Turbine Area Ventilation System	All EPUs except where the application demonstrates that previous analysis is bounding	SPSB		9.4.4 Rev. 2 July 1981	GDC-60		2.7.5	2.7.5	

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review Checklist
							BWR	PWR	
ESF Ventilation System	All EPUs except where the application demonstrates that previous analysis is bounding	SPSB		9.4.5 Rev. 2 July 1981	GDC-4 GDC-17 GDC-60		2.7.6	2.7.6	

Notes:

1. Under SRP Section 6.4, Section II, "Acceptance Criteria," the discussion for Item C related to GDC-19 should be supplemented with "and providing a suitably controlled environment for the control room operators and the equipment located therein."

2. Under SRP Section 6.4, Section II, Item 2, "Ventilation System Criteria," the discussion related to review of the control room area ventilation system under SRP Section 9.4.1 should be retained.

LIST OF ACRONYMS FOR MATRIX 7

BWR = boiling-water reactor
EPUs = extended power uprates
ESF = engineered safety feature
GDC = General Design Criterion
PWR = pressurized-water reactor
SPSB = Probabalistic Safety Assessment Branch
SRP = Standard Review Plan

ATTACHMENT 1 TO MATRIX 7

Independent Calculations

Habitability, Filtration, and Ventilation

Use the criteria in the Standard Review Plan sections referenced in Matrix 7 for determining when to perform independent calculations.

MATRIX 8

SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

Reactor Systems

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review Checklist
							BWR	PWR	
Fuel System Design	All EPUs	SRXB		4.2 Draft Rev. 3 April 1996	10 CFR 50.46 GDC-10 GDC-27 GDC-35	Note 1* Note 2*	2.8.1	2.8.1	
Nuclear Design	All EPUs	SRXB		4.3 Draft Rev. 3 April 1996	GDC-10 GDC-11 GDC-12 GDC-13 GDC-20 GDC-25 GDC-26 GDC-27 GDC-28	RG 1.190 GSI 170 IN 97-085	2.8.2	2.8.2	
Thermal and Hydraulic Design	All EPUs	SRXB		4.4 Draft Rev. 2 April 1996	GDC-10 GDC-12	Note 3*	2.8.3	2.8.3	

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number BWR	Template Safety Evaluation Section Number PWR	Acceptance Review Checklist
Functional Design of Control Rod Drive System	All EPUs	SRXB	SPLB	4.6 Draft Rev. 2 April 1996	GDC-4 GDC-23 GDC-25 GDC-26 GDC-27 GDC-28 GDC-29 10 CFR 50.62(c)(3)		2.8.4.1	2.8.4.1	
Overpressure Protection during Power Operation	All EPUs	SRXB		5.2.2 Draft Rev. 3 April 1996	GDC-15 GDC-31	Note 4*	2.8.4.2	2.8.4.2	
Overpressure Protection during Low Temperature Operation	PWR EPUs	SRXB		5.2.2 Draft Rev. 3 April 1996	GDC-15 GDC-31			2.8.4.3	
Reactor Core Isolation Cooling System	BWR EPUs	SRXB		5.4.6 Draft Rev. 4 April 1996	GDC-4 GDC-5 GDC-29 GDC-33 GDC-34 GDC-54 10 CFR 50.63		2.8.4.3		
Residual Heat Removal System	All EPUs	SRXB		5.4.7 Draft Rev. 4 April 1996	GDC-4 GDC-5 GDC-19 GDC-34	Note 5*	2.8.4.4	2.8.4.4	
Emergency Core Cooling System	All EPUs	SRXB		6.3 Draft Rev. 3 April 1996	GDC-4 GDC-27 GDC-35 10 CFR 50.46 10 CFR 50 App. K	Note 6*	2.8.5.6.2	2.8.5.6.3	

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review Checklist
							BWR	PWR	
Standby Liquid Control System	BWR EPUs	SRXB	EMCB SPLB	9.3.5 Draft Rev. 3 April 1996	GDC-26 GDC-27 10 CFR 50.62(c)(4)	Note 12*	2.8.4.5		
Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve	All EPUs	SRXB		15.1.1-4 Draft Rev. 2 April 1996	GDC-10 GDC-15 GDC-20 GDC-26	Note 7*	2.8.5.1	2.8.5.1.1	
Steam System Piping Failures Inside and Outside of Containment	PWR EPUs	SRXB		15.1.5 Draft Rev. 3 April 1996	GDC-27 GDC-28 GDC-31 GDC-35	Note 7*		2.8.5.1.2	
Loss of External Load; Turbine Trip, Loss of Condenser Vacuum; Closure of Main Steam Isolation Valve (BWR); and Steam Pressure Regulator Failure (Closed)	All EPUs	SRXB		15.2.1-5 Draft Rev. 2 April 1996	GDC-10 GDC-15 GDC-26	Note 7*	2.8.5.2.1	2.8.5.2.1	
Loss of Nonemergency AC Power to the Station Auxiliaries	All EPUs	SRXB		15.2.6 Draft Rev. 2 April 1996	GDC-10 GDC-15 GDC-26	Note 7*	2.8.5.2.2	2.8.5.2.2	
Loss of Normal Feedwater Flow	All EPUs	SRXB	EEIB	15.2.7 Draft Rev. 2 April 1996	GDC-10 GDC-15 GDC-26	Note 7*	2.8.5.2.3	2.8.5.2.3	
Feedwater System Pipe Breaks Inside and Outside Containment	PWR EPUs	SRXB	EEIB	15.2.8 Draft Rev. 2 April 1996	GDC-27 GDC-28 GDC-31 GDC-35	Note 7*		2.8.5.2.4	

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review Checklist
							BWR	PWR	
Loss of Forced Reactor Coolant Flow Including Trip of Pump Motor and Flow Controller Malfunctions	All EPUs	SRXB		15.3.1-2 Draft Rev. 2 April 1996	GDC-10 GDC-15 GDC-26	Note 7*	2.8.5.3.1	2.8.5.3.1	
Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break	All EPUs	SRXB		15.3.3-4 Draft Rev. 3 April 1996	GDC-27 GDC-28 GDC-31	Note 7*	2.8.5.3.2	2.8.5.3.2	
Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition	All EPUs	SRXB		15.4.1 Draft Rev. 3 April 1996	GDC-10 GDC-20 GDC-25	Note 7*	2.8.5.4.1	2.8.5.4.1	
Uncontrolled Control Rod Assembly Withdrawal at Power	All EPUs	SRXB		15.4.2 Draft Rev. 3 April 1996	GDC-10 GDC-20 GDC-25	Note 7*	2.8.5.4.2	2.8.5.4.2	
Control Rod Misoperation (System Malfunction or Operator Error)	PWR EPUs	SRXB		15.4.3 Draft Rev. 3 April 1996	GDC-10 GDC-20 GDC-25	Note 7*		2.8.5.4.3	
Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate	All EPUs	SRXB		15.4.4-5 Draft Rev. 2 April 1996	GDC-10 GDC-15 GDC-20 GDC-26 GDC-28	Note 7*	2.8.5.4.3	2.8.5.4.4	
Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant	PWR EPUs	SRXB		15.4.6 Draft Rev. 2 April 1996	GDC-10 GDC-15 GDC-26	Note 7*		2.8.5.4.5	

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review Checklist
							BWR	PWR	
Spectrum of Rod Ejection Accidents	PWR EPUs	SRXB		15.4.8 Draft Rev. 3 April 1996	GDC-28	Note 7*		2.8.5.4.6	
Spectrum of Rod Drop Accidents	BWR EPUs	SRXB		15.4.9 Draft Rev. 3 April 1996	GDC-28	Note 7*	2.8.5.4.4		
Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory	All EPUs	SRXB		15.5.1-2 Draft Rev. 2 April 1996	GDC-10 GDC-15 GDC-26	Note 7* Note 8*	2.8.5.5	2.8.5.5	
Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve or a BWR Pressure Relief Valve	All EPUs	SRXB		15.6.1 Draft Rev. 2 April 1996	GDC-10 GDC-15 GDC-26	Note 7*	2.8.5.6.1	2.8.5.6.1	
Steam Generator Tube Rupture	PWR EPUs	SRXB		15.6.3 Draft Rev. 3 April 1996	Note 7*	Note 7*		2.8.5.6.2	
Loss-of Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary	All EPUs	SRXB		15.6.5 Draft Rev. 3 April 1996	GDC-35 10 CFR 50.46	Note 7* Note 9*	2.8.5.6.2	2.8.5.6.3	
Anticipated Transient Without Scram	All EPUs	SRXB				Note 7* Note 10*	2.8.5.7	2.8.5.7	
New Fuel Storage	EPU applications that request approval for new fuel.	SRXB		9.1.1 Draft Rev. 3 April 1996	GDC-62		2.8.6.1	2.8.6.1	

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review Checklist
							BWR	PWR	
Spent Fuel Storage	EPU applications that request approval for new fuel.	SRXB		9.1.2 Draft Rev. 4 April 1996	GDC-4 GDC-62		2.8.6.2	2.8.6.2	

Notes:

1. When mixed cores (i.e., fuels of different designs) are used, the review covers the licensee's evaluation of the effects of mixed cores on design-basis accident and transient analyses.

2. The current acceptance criteria for fuel damage for reactivity insertion accidents (RIAs) requires revision per Research Information Letter No. 174, "Interim Assessment of Criteria for Analyzing Reactivity Accidents at High Burnup." The Office of Nuclear Regulatory Research is conducting confirmatory research on RIAs and the Office of Nuclear Reactor Regulation is discussing the issue of fuel damage criteria with the nuclear power industry as part of the industry's proposal to increase fuel burnup limits in the future. In the interim, current methods for assessing fuel damage in RIAs are considered acceptable based on the NRC staff's understanding of actual fuel performance, as shown in three-dimensional kinetic calculations which indicate acceptably low fuel cladding enthalpy.

3. The review also covers core design changes and any effects on radial and bundle power distribution, including any changes in critical heat flux ratio and critical power ratio. The review will also confirm the adequacy of the flow-based average power range monitor flux trip and safety limit minimum critical power ratio at the uprated conditions.

4. The review also covers the method used in determining allowable power levels with inoperable main steam safety valves.

5. The review also covers the total time necessary to reach the shutdown cooling initiation temperature.

6. The review for BWRs will cover the justification for changes in calculated peak cladding temperature (PCT) for the licensing-basis case and the upper-bound case and any impact of the changes in PCTs on the use of the licensing methods for the power uprate.

7. The review also confirms:
   - The licensee used NRC-approved codes and methods for the plant-specific application and the licensee's use of the codes and methods complies with any limitations, restrictions, and conditions specified in the approving safety evaluation.
   - All changes of reactor protection system trip delays are correctly addressed and accounted for in the analyses.
   - (For PWRs) Steam generator plugging and asymmetry limits are accounted for in the analyses.
   - (For PWRs) The licensee's evaluation of the effects of Westinghouse Nuclear Service Advisory Letters (NSALs), NSAL 02-3 and Revision 1, NSAL 02-4, and NSAL 02-5. These NSALs document problems with water level setpoint uncertainties in Westinghouse-designed steam generators. The review is conducted to ensure that the effects of the problems identified have been accounted for in steam generator water level setpoints used in LOCA, non-LOCA, and ATWS analyses.
8. For the inadvertent operation of emergency core cooling system and chemical and volume control system malfunctions that increases reactor coolant inventory events: (a) non-safety-grade pressure-operated relief valves should not be credited for event mitigation and (b) pressurizer level should not be allowed to reach a pressurizer water-solid condition.

9. The review also verifies that:
   - Licensee and vendor processes ensure LOCA analysis input values for PCT-sensitive parameters bound the as-operated plant values for those parameters
   - (For PWRs) The models and procedures continue to comply with 10 CFR 50.46 during the switchover from the refueling water storage tank to the containment sump (i.e., the core remains adequately cool during any flow reduction or interruption that may occur during switchover).
   - (For PWRs) Large-break LOCA analyses account for boric acid buildup during long-term core cooling and that the predicted time to initiate hot leg injection is consistent with the times in the operating procedures.
   - (For BWRs) The licensee's comparison of parameters used in the LOCA analysis with actual core design parameters provide the needed justification to confirm the applicability of the generic LOCA methodology.

10. The ATWS review is conducted to ensure that the plant meets the 10 CFR 50.62 requirements:

   - For PWR plants with both a diverse scram system (DSS) and ATWS mitigation system actuation circuitry (AMSAC), the staff will not review ATWS for EPUs.
   - For PWR plants where a DSS is not specifically required by 10 CFR 50.62, a review is conducted to verify that the consequences of an ATWS are acceptable. The acceptance criteria is that the peak primary system pressure should not exceed the ASME Service Level C limit of 3200 psig. The peak ATWS pressure is primarily a function of the moderator temperature coefficient and the primary system relief capacity.
   - For BWR plants, the review is conducted to ensure that the licensee has appropriately accounted for changes in analyses due to the uprated power level and confirm that required equipment, such as the standby liquid control system (SLCS) pumps, can deliver required flowrates. The review will also cover the SLCS relief valve margin. In addition, a review is conducted to ensure that SLCS flow can be injected at the assumed time without lifting bypass relief valves during the limiting ATWS.

LIST OF ACRONYMS FOR MATRIX 8

BWR = boilling-water reactor
CFR = *Code of Federal Regulations*
EMCB = Materials and Chemical Engineering Branch
EPUs = extended power uprates
GDC = general design criterion
PWR = pressurized-water reactor
SPLB = Plant Systems Branch
SRP = standard review plan
SRXB = Reactor Systems Branch
PWR = pressurized-water reactor
SPLB = Plant Systems Branch
EMCB = Materials & Chemical Engineering Branch
LOCA = loss-of-coolant accident
ATWS = anticipated transients without scram
ASME = American Society of Mechanical Engineers
AMSAC = ATWS Mitigation System Actuation Circuitry
DSS = Diverse Scram System

ATTACHMENT 1 TO MATRIX 8

Independent Calculations

Reactor Systems

Use the following guidelines for determining when to perform independent calculations:

- The licensee has performed analyses that included deviations that have not been previously approved by the NRC staff for the plant under review, or for a similar plant at similar power levels or power densities
- The licensee has performed analyses using a methodology that is questionable or has not been previously used at the plant, or at a similar plant at similar power levels.
- The licensee's analyses incorporate substantial changes to methodologies used in previously approved analyses.
- The licensee's analyses extend the range of applicability of the methodologies beyond previously approved limits.
- The licensee has performed analyses using first-of-a-kind methodologies.
- The licensee has performed analyses using assumptions that are questionable or which have changed substantially since they were approved or used in plants operating at similar power levels.
- The licensee has not adequately addressed the impact of the proposed extended power uprate on the assumptions, range of applicability, or suitability of the methods used for the analyses.
- The results of the licensee's analyses are questionable in light of (1) the results of other similar NRC staff review experience, (2) the results of other NRC staff calculations, (3) the results of ongoing research activities, or (4) the results of operating experience.
- The licensee's analyses show significant reductions in available margin to minimally acceptable levels.

Following are examples of the types of analyses that SRXB may perform in support of power uprates. Additional detailed examples are under development.

Independent Analysis Criteria - BWR Licensing Actions

	LOCA	Transient	Core Design	Sub-Channel	RIA Events	ATWS	Stability
Power uprate beyond previously performed	x	x		x	x	x	x
Change in Fuel Vendor	x		x		x		x
Use of new fuel design	x	x	x			x	x
MELLLA Implementation						x	x
Increase in Power density >10%	x	x				x	x
Increase in peaking factor	x						
SLMCPR Change >0.03	x	x				x	x
ECCS temperature change >50 °F	x						

MATRIX 9

SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

Source Terms and Radiological Consequences Analyses

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review Checklist
							BWR	PWR	
Source Terms for Input into Radwaste Management Systems Analyses	All EPUs	SPSB		11.1 Draft Rev. 3 April 1996	10 CFR 20 10 CFR 50, App. I GDC-60		2.9.1	2.9.1	
Radiological Consequence Analyses Using Alternative Source Terms	EPUs that utilize alternative source term	SPSB	EEIB EMCB EMEB IEPB SPLB SRXB	15.0.1 Rev. 0 July 2000	10 CFR 50.67 GDC-19 10 CFR 50.49 10 CFR 51 10 CFR 50, App. E NUREG-0737		2.9.2	2.9.2	
Radiological Consequences of Main Steamline Failures Outside Containment for a PWR	PWR EPUs that do not utilize alternative source term whose main steamline break analyses result in fuel failure	SPSB	SRXB	15.1.5, App. A Draft Rev. 3 April 1996	10 CFR 100	Notes 4, 5, 6, 7, 27*		2.9.2	
				6.4 Draft Rev. 3 April 1996	GDC-19	Notes 1, 2, 3, 28, 29*			
Radiological Consequences of Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break	EPUs that do not utilize alternative source term whose reactor coolant pump rotor seizure or reactor coolant pump shaft break results in fuel failure	SPSB	SRXB	15.3.3-4 Draft Rev. 3 April 1996	10 CFR 100	Notes 5, 8, 9, 27*		2.9.3	
				6.4 Draft Rev. 3 April 1996	GDC-19	Notes 1, 2, 3, 28, 29*			

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review Checklist
							BWR	PWR	
Radiological Consequences of a Control Rod Ejection Accident	PWR EPUs that do not utilize alternative source term whose rod ejection accident results in fuel failure or melting	SPSB	SRXB	15.4.8, App. A Draft Rev. 2 April 1996	10 CFR 100	Notes 4, 21, 22, 27*		2.9.4	
				6.4 Draft Rev. 3 April 1996	GDC-19	Notes 1, 2, 3, 28, 29*			
Radiological Consequences of Control Rod Drop Accident	BWR EPUs that do not utilize alternative source term whose control rod drop accident results in fuel failure or melting	SPSB	SRXB	15.4.9, App. A Draft Rev. 3 April 1996	10 CFR 100	Notes 9, 10, 27*	2.9.2		
				6.4 Draft Rev. 3 April 1996	GDC-19	Notes 1, 2, 3, 28, 29*			
Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment	EPUs that do not utilize alternative source term whose failure of small lines carrying primary coolant outside containment result in fuel failure	SPSB		15.6.2 Draft Rev. 3 April 1996	GDC-55 10 CFR 100		2.9.3	2.9.5	
				6.4 Draft Rev. 3 April 1996	GDC-19	Notes 1, 2, 3, 28, 29*			

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review Checklist
							BWR	PWR	
Radiological Consequences of Steam Generator Tube Failure	PWR EPUs that do not utilize alternative source term whose steam generator tube failure results in fuel failure	SPSB	SRXB	15.6.3 Draft Rev. 3 April 1996	10 CFR 100	Notes 4, 13, 14, 15, 27*		2.9.6	
				6.4 Draft Rev. 3 April 1996	GDC-19	Notes 1, 2, 3, 28, 29*			
Radiological Consequences of Main Steamline Failure Outside Containment for a BWR	BWR EPUs that do not utilize alternative source term whose main steam line failure outside containment results in fuel failure	SPSB	SRXB	15.6.4 Draft Rev. 3 April 1996	10 CFR 100	Note 27*	2.9.4		
				6.4 Draft Rev. 3 April 1996	GDC-19	Notes 1, 2, 3, 28, 29*			
Radiological Consequences of a Design Basis Loss-Of-Coolant-Accident Including Containment Leakage Contribution	EPUs that do not utilize alternative source term	SPSB	SPLB	15.6.5, App. A Draft Rev. 2 April 1996	10 CFR 100	Notes 4, 23, 24, 25, 26, 27*	2.9.5	2.9.7	
				6.4 Draft Rev. 3 April 1996	GDC-19	Notes 1, 2, 3, 28, 29*			

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review Checklist
							BWR	PWR	
Radiological Consequences of a Design Basis Loss-Of-Coolant-Accident: Leakage from ESF Components Outside Containment	EPUs that do not utilize alternative source term	SPSB	SPLB	15.6.5, App. B Draft Rev. 2 April 1996	10 CFR 100	Notes 11, 27*	2.9.5	2.9.7	
				6.4 Draft Rev. 3 April 1996	GDC-19	Notes 1, 2, 3, 28, 29*			
Radiological Consequences of a Design Basis Loss-Of-Coolant-Accident: Leakage from Main Steam Isolation Valves	BWR EPUs that do not utilize alternative source term	SPSB		15.6.5, App. D Draft Rev. 2 April 1996	10 CFR 100	Notes 9, 12, 27*	2.9.5		
				6.4 Draft Rev. 3 April 1996	GDC-19	Notes 1, 2, 3, 28, 29*			
Radiological Consequences of Fuel Handling Accidents	EPUs that do not utilize alternative source term	SPSB	SPLB	15.7.4 Draft Rev. 2 April 1996	10 CFR 100 GDC-61	Notes 4, 5, 18, 19, 20, 27*	2.9.6	2.9.8	
				6.4 Draft Rev. 3 April 1996	GDC-19	Notes 1, 2, 3, 28, 29*			
Radiological Consequences of Spent Fuel Cask Drop Accidents	EPUs that do not utilize alternative source term	SPSB	EMEB SPLB	15.7.5 Draft Rev. 3 April 1996	10 CFR 100 GDC-61	Notes, 5, 16, 17, 8, 18, 27*	2.9.7	2.9.9	
				6.4 Draft Rev. 3 April 1996	GDC-19	Notes 1, 2, 3, 28, 29*			

Notes:

1. In addition to SRP Section 15.6.5, Appendices A, B, and D, dose consequences in the control room are determined from design-basis accidents as part of the review for SRP Sections 15.0.1; 15.1.5, Appendix A; 15.3.3-4, 15.4.8, Appendix A; 15.4.9, Appendix A; 15.6.2, 15.6.3, 15.6.4, 15.7.4, and 15.7.5.

2. Regulatory Guide 1.95 was canceled. Relevant guidance from Regulatory Guide 1.95 was incorporated into Regulatory Guide 1.78, Revision 1 in January 2002. Therefore, Regulatory Guide 1.95 should not be used.

3. Table 6.4-1, attached to SRP Section 6.4 and referred to in Item 7, "Independent Analyses," of the "Review Procedures" Section of SRP Section 6.4 may not be used.

4. Acceptable dose conversion factors may be taken from Table 2.1 of Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," Environmental Protection Agency, 1988; and Table III.1 of Federal Guidance Report 12, " External Exposure to Radionuclides in Air, Water, and Soil," Environmental Protection Agency, 1993.

5. NUREG-1465 should not be used.

6. For the review of the main steamline failure accident, review of facilities licensed with, or applying for, alternative repair criteria (ARC) should use SRP Section 15.1.5, Appendix A, in conjunction with the guidance in Draft Regulatory Guide DG-1074, "Steam Generator Tube Integrity," December 1998, for acceptable assumptions and methodologies for performing radiological analyses.

7. For facilities that implement ARC, the primary-to-secondary leak rate in the faulted generator should be assumed to be the maximum accident-induced leakage derived from the repair criteria and burst correlations. The leak rate limiting condition for operation specified in the technical specifications is equally apportioned among the unaffected steam generators.

8. Guidance for the radiological consequences analyses review with respect to acceptable modeling of the radioactivity transport is given in SRP Section 15.6.3, "Radiological Consequences of Steam Generator Tube Failure (PWR)," for applicants that use the traditional source term, based on TID-14844.

9. References to specific computer codes (e.g., SARA, TACT, Pipe Model) are not necessary since other computer codes/methods may be used.

10. In the second paragraph of Section III, "Review Procedure," it is stated that the control rod drop accident is expected to result in radiological consequences less than 10 percent of the 10 CFR Part 100 guideline values, even with conservative assumptions. The value of 10 percent should be replaced with 25 percent.

11. In Section III, "Review Procedures," the guidance in the fourth paragraph, which deals with passive failures, should not be used.

12. The last paragraph on page 15.6.5-4 refers to a "code" developed by J. E. Cline and Associates, Inc. This is identified as Reference 5 in the paragraph. The word "code" should be changed to "model" because the staff does not have the computer code. In addition, the correct reference to the work by J. E. Cline and Associates, Inc., is 4.

13. Item 4 of the "Review Interfaces" section should be deleted. SPSB review of the steam generator tube rupture accidents for their contribution to plant risk is not currently used in the design-basis accident review for radiological consequences.

14. The reference to Figure 3.4-1 of the Nuclear Steam Supply System vendor Standard Technical Specification in Item 6.(a) of Section III, "Review Procedures," does not apply. In addition, the primary coolant iodine concentration discussed in this Item is the 48-hour maximum value.

15. In Item 6.(b) of Section III, "Review Procedures," the multiplier of 500 used for estimating the increase in iodine release rate is reduced to 335 as a result of the staff's review of iodine release rate data collected by Adams and Atwood.

16. The reference to SRP Section 9.1.4 in Item 2.c of the "Review Interfaces" section should be changed to SRP Section 9.1.5.

17. The reference to Regulatory Guide 1.25, which was deleted in 1996, should be retained, with exceptions as noted below in Note 18.

18. The following exceptions to Regulatory Guide 1.25 are provided. These exceptions are based on the staff's review of NUREG/CR-6703.

The fraction of the core inventory assumed to be in the gap for the various nuclides are given in the table below. The release fractions from the table are used in conjunction with the calculated fission product inventory and the maximum core radial peaking factor. These release fractions have been determined to be acceptable for use with currently approved LWR fuel with a peak burnup up to 62,000 MWD/MTU, provided that the maximum linear heat generation rate will not exceed 6.3 kW/ft peak rod average power for rods with burnups that exceed 54 GWD/MTU. As an alternative, fission gas release calculations using NRC-approved methodologies may be considered on a case-by-case basis.

NON-LOCA FRACTION OF FISSION PRODUCT INVENTORY IN GAP	
GROUP	**FRACTION**
I-131	0.08
Kr-85	0.10
Other Noble Gases	0.05
Other Iodines	0.05

19. References to the Standard Technical Specifications should be replaced with references to the plant-specific technical specifications or technical requirements manual (TRM).

20. Technical Specification Task Force (TSTF) Traveler TSTF-51 proposed to add the term "recently" to the applicability section of certain technical specifications. The proposed change in intended to remove certain technical specifications requirements for operability of ESF systems (e.g., secondary containment isolation and filtration systems) during refueling. The associated technical specifications bases define "recently" as the minimum decay time used in supporting radiological consequences analyses of fuel handling accidents. Radiological consequences analyses for these applicants should generally assume a 2-hour release directly to the environment, without holdup or mitigation by ESF systems and no credit for containment closure. Additionally, licensees adding the term "recently" must make a commitment for a single normal or contingency method to promptly close primary or secondary containment penetrations. Such prompt methods need not completely block the penetration or be capable of resisting pressure. The review of this commitment and the prompt methods should be coordinated with IORB, SPLB, and IEPB.

21. In the last sentence of Item 2 of the "Review Interfaces" section, the reference to the number of fuel pins experiencing departure from nucleate boiling (DNB) should be deleted. The reference to fuel clad melting should be used and is therefore retained.

22. In Item 2 of the "Review Procedures" section, the references to the "number of fuel pins reaching DNB" should be deleted and replaced with "the number of fuel pins with cladding failure." In addition, the use of a conservative value of 10 percent for fuel cladding failure in the calculation of the radiological consequences of the rod ejection accident is acceptable.

23. In Item 1 of the "Areas of Review" section, the use of the word "established" is incorrect. The word "established" should be replaced with the word "assessed."

24. In Item 1 of the "Acceptance Criteria" section, the following text in the last line should be deleted: "3.0 Sv (300 rem) to the thyroid and 0.25 Sv (25 rem) to the whole body."

25. In Item 1 of the "Review Procedures" section, the following should be added after the first sentence:

    Appendix K to 10 CFR Part 50 defines conservative analysis assumptions for evaluation of ECCS performance during design-basis LOCAs. Appendix K requires the licensees to assume that the reactor has been operating continuously at a power level at least 1.02 times the licensed power level to allow for instrumentation error. Appendix K allows for an assumed power level less than 1.02 times the licensed power level but not less than the licensed power level, provided the alternative value has been demonstrated to account for uncertainties due to power level instrumentation error.

26. In Item 2 of the "Review Procedures" section, the following statements should be deleted:

    "A check is made of the LOCA [loss-of-coolant accident] assumptions listed in Chapter 15 of the SAR to verify that the primary containment leakage rate has been assumed to remain constant over the course of the accident for a BWR and to remain constant at one half of the initial leak rate after 24 hours for a PWR."

    "The leakage rate used should correspond to that given in the technical specification."

The above statements should be replaced with the following:

    "A check is made of the LOCA assumptions listed in Chapter 15 of the SAR to verify acceptable primary containment leakage assumptions. The primary containment should be assumed to leak at the peak pressure technical specification leak rate for the first 24 hours. For PWRs, the leakage rate may be reduced after the first 24 hours to 50 percent of the TS leak rate. For BWRs, leakage may be reduced after the first 24 hours, if supported by plant configuration and analyses, to a value not less than 50 percent of the TS leak rate. Leakage from subatmospheric containments is assumed to terminate when the containment is brought to and maintained at a subatmospheric condition, as defined by the TSs."

27. The staff has drafted updated guidance on performing design-basis radiological analyses in draft Regulatory Guide DG-1113, "Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors," issued for public comment January 2002. The resulting final regulatory guide may be used for guidance on review of design-basis accident non-alternative source term radiological analyses after the date of issuance of the final regulatory guide.

28. In Section II, "Acceptance Criteria," the discussion for Item C related to GDC-19 should be supplemented with

    "and providing a suitably controlled environment for the control room operators and the equipment located therein."

29. In Section II, Item 2, "Ventilation System Criteria," the discussion related to review of the control room area ventilation system under SRP Section 9.4.1 should be retained.

LIST OF ACRONYMS FOR MATRIX 9

BWR = boiling-water reactor
CFR = *Code of Federal Regulations*
EEIB = Electrical & Instrumentation & Controls Branch
EMCB = Materials & Chemical Engineering Branch
EMEB = Mechanical & Civil Engineering Branch
EPUs = extended power uprates
GDC = General Design Criterion
IEPB = Emergency Preparedness and Plant Support Branch
PWR = pressurized-water reactor
IROB = Reactor Operations Branch
SPLB = Plant Systems Branch
SPSB = Probabalistic Safety Assessment Branch
SRP = Standard Review Plan
SRXB = Reactor Systems Branch

ATTACHMENT 1 TO MATRIX 9

Independent Calculations

Radiological Consequences Analyses

Use the following guidelines for determining when to perform independent calculations:

- The licensee performed analyses that included deviations that have not been previously approved by the NRC staff for the plant.
- The licensee performed analyses using a methodology that is questionable or has not been previously used at the plant.
- The licensee's analyses incorporate substantial changes to methodologies used in previous analyses.
- The licensee performed analyses using first-of-a-kind methodologies.
- The licensee performed analyses using assumptions that are questionable or contained substantial changes.
- The licensee has not adequately addressed the impact of the proposed extended power uprate on assumptions or methods used in the analyses.
- The results of the licensee's analyses are questionable.

MATRIX 10

SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

Health Physics

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review Checklist
							BWR	PWR	
Radiation Sources	All EPUs	IEPB		12.2 Draft Rev. 3 April 1996	10 CFR 20		2.10.1	2.10.1	
Radiation Protection Design Features	All EPUs	IEPB		12.3-4 Draft Rev. 3 April 1996	10 CFR 20 GDC-19	Note 1*	2.10.1	2.10.1	
Operational Radiation Protection Program	All EPUs	IEPB		12.5 Draft Rev. 3 April 1996	10 CFR 20	Note 2* Note 3*	2.10.1	2.10.1	

Notes:

1. Regulatory Guide 8.12, "Criticality Accident Alarm Systems" has been withdrawn and should not be used.

2. Regulatory Guide 8.3, "Film Badge Performance Criteria" has been withdrawn and should not be used.

3. Regulatory Guide 8.14, "Personnel Neutron Dosimeters" has been withdrawn and should not be used.

LIST OF ACRONYMS FOR MATRIX 10

BWR = boiling-water reactor
CFR = *Code of Federal Regulations*
EPUs = extended power uprates
GDC = General Design Criterion
IEPB = Emergency Preparedness and Plant Support Branch
PWR = pressurized-water reactor
SRP = Standard Review Plan

ATTACHMENT 1 TO MATRIX 10

Independent Calculations

Health Physics

Independent calculations are not performed in the area of health physics. The primary area of concern related to health physics with respect to extended power uprates is the effect of these power increases on plant dose rates and the adequacy of plant shielding. However, past experience with extended power uprate reviews has shown that extended power uprates up to 20 percent have little effect on plant dose rates in most areas of the plant due to the built-in conservatism designed into the plant's shielding.

MATRIX 11

SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

Human Performance

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review Checklist
							BWR	PWR	
Reactor Operator Training	All EPUs	IROB		13.2.1 Draft Rev. 2 Dec. 2002	Specific review questions are provided in the template safety evaluations.		2.11	2.11	
Training for Non-Licensed Plant Staff	All EPUs	IROB		13.2.2 Draft Rev. 2 Dec. 2002	Specific review questions are provided in the template safety evaluations.		2.11	2.11	
Operating and Emergency Operating Procedures	All EPUs	IROB	SPLB SPSB SRXB	13.5.2.1 Draft Rev. 1 Dec. 2002	Specific review questions are provided in the template safety evaluations.		2.11	2.11	
Human Factors Engineering	All EPUs	IROB		18.0 Draft Rev. 1 Dec. 2002	Specific review questions are provided in the template safety evaluations.		2.11	2.11	

LIST OF ACRONYMS FOR MATRIX 11

BWR = boiling-water reactor
EPUs = extended power uprates
IROB = Reactor Operations Branch
PWR = pressurized-water reactor
SPLB = Plant Systems Branch
SPSB = Probabilistic Safety Assessment Branch
SRP = Standard Review Plan
SRXB = Reactor Systems Branch

ATTACHMENT 1 TO MATRIX 11

Independent Calculations

Human Performance

Perform an independent calculation of operator available response time based on the criteria of ANSI/ANS-58.8, "Time Response Design Criteria for Safety-Related Operator Actions," for those operator actions for which the EPU has caused a reduction in available action time. This independent calculation is to be used for screening purposes only. Should the calculation indicate a timing issue, request the licensee to demonstrate, through simulation or other means, that operators can successfully perform the action in the available time.

MATRIX 12

SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

Power Ascension and Testing Plan

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review Checklist
							BWR	PWR	
Power Ascension and Testing	All EPUs	IEPB	EEIB EMCB EMEB IROB SPLB SPSB SRXB	14.2.1 Rev. 0 XXXX 2003	Entire Section		2.12	2.12	

LIST OF ACRONYMS FOR MATRIX 12

BWR = boiling-water reactor
EEIB = Electrical & Instrumentation & Controls Branch
EMCB = Materials and Chemical Engineering Branch
EMEB = Mechanical & Civil Engineering Branch
EPUs = extended power uprates
IEPB = Emergency Preparedness and Plant Support Branch
IROB = Reactor Operations Branch
PWR = pressurized-water reactor
SPLB = Plant Systems Branch
SPSB = Probabalistic Safety Assessment Branch
SRP = Standard Review Plan
SRXB = Reactor Systems Branch

ATTACHMENT 1 TO MATRIX 12

Independent Calculations

Power Ascension and Testing Plan

The review of the power ascension and testing plan for extended power uprates is based on technical reviews of other areas. Independent calculations for those areas are identified in the attachments to their respective matrices.

MATRIX 13

SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

Risk Evaluation

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review Checklist
							BWR	PWR	
Risk Evaluation	All EPUs	SPSB				Note 1* RG 1.174 RIS 2001-02	2.13	2.13	

Notes:
1. The staff's review is based on Attachment 2 to this matrix. Attachment 2 invokes SRP Chapter 19, Appendix D, if special circumstances are identified during the review.

LIST OF ACRONYMS FOR MATRIX 13

BWR = boiling-water reactor
EPUs = extended power uprates
PWR = pressurized-water reactor
RG = regulatory guide
RIS = regulatory issue summary
SPSB = Probabalistic Safety Assessment Branch
SRP = Standard Review Plan

ATTACHMENT 1 TO MATRIX 13

Independent Calculations

Risk Evaluation

Use the guidance in Attachment 2 to Matrix 13 of RS-001 for determining when to perform independent calculations.

ATTACHMENT 2 TO MATRIX 13

Supplemental Risk Evaluation Review Guidance

Risk Evaluation

1. INTRODUCTION

In addition to ensuring that a license amendment request complies with the U.S. Nuclear Regulatory Commission's (NRC's) regulations and other requirements, it is also the staff's responsibility to consider the risk aspects of a license amendment request (cf. COMSAJ-97-08 and RIS 2001-02). The use of risk information is clear when the licensee or the NRC designates the submittal as a "risk-informed" license application. Guidance is also provided to the staff in Appendix D of Chapter 19 of the Standard Review Plan (SRP) (Reference 1) as to the "special circumstances" under which a detailed risk review may be required, even for license applications that are not designated as being risk-informed. This process is also described in Regulatory Issue Summary (RIS) 2001-02 (Reference 2). Special circumstances is defined in the above guidance as "conditions or situations that would raise questions about whether there is adequate protection, and that could rebut the normal presumption of adequate protection from compliance with existing requirements. In such situations, undue risk may exist even when all regulatory requirements are satisfied."

Though power uprates are not submitted as risk-informed license applications, it is recognized that there are potential risk increases associated with implementing a power uprate due to the increased heat loads at higher powers and the resulting reductions in the times available to perform specific accident response actions. In addition, there can be impacts on the equipment loads and the potential for an increase in the frequency of reactor scrams due to these increased loads and tighter operating margins. For small power uprates (i.e., those referred to as measurement uncertainty recapture power uprates and stretch power uprates), the risk increases are expected to be exceedingly small. However, notwithstanding any plant modifications that could reduce risks, some increase in risk is expected for larger power uprates. Depending on the type of plant-specific modifications required to implement the larger power uprates, these power uprates have the potential for significantly increasing plant risks, especially if they significantly impact initiating event frequencies, component reliabilities, system success criteria, and/or operator response times. Further, large power uprate requests are specifically identified in Appendix D to SRP Chapter 19 as an example of the type of situation that might create "special circumstances" since they could "involve changes for which the synergistic or cumulative effects could significantly impact risk." Therefore, the Probabilistic Safety Assessment Branch (SPSB) Safety Program Section formally reviews all license application submittals for extended power uprates.

As of December 2002, the SPSB Safety Program Section staff had performed risk reviews of eight extended power uprate license applications involving twelve units. All, but one, of these applications were for boiling water reactors (BWRs) of various design vintages, including: five BWR-3/Mark-I units (Monticello, Dresden 2 and 3, and Quad Cities 1 and 2), five BWR-4/Mark-I units (Hatch 1 and 2, Duane Arnold, and Brunswick 1 and 2), and one BWR-6/Mark III unit (Clinton). The one pressurized water reactor (PWR) extended power uprate license application was for a Combustion Engineering (CE) plant with a large dry containment (Arkansas Nuclear One - Unit 2). The extended power uprates have been as high as 20 percent of OLTP.

The staff, recognizing the need to address the potential risk increase associated with extended power uprates, stated in a 1996 position paper (Reference 3) that licensees should conduct risk evaluations for extended power uprate license applications. Specifically, the paper states that it is appropriate for each applicant to assess the effect of the proposed power uprate on the results of its independent plant examination (IPE)/probabilistic risk assessment (PRA) and that this assessment should cover the potential impacts on initiating event frequencies, success criteria, component failure rates, and the time available for operator actions and equipment restoration. The paper also states that these inputs and assumptions are examples of the appropriate areas of the IPE/PRA for review and expects that applicants will address any other areas that the applicants determine also may be affected by power uprate. Finally, the paper states that the staff will request that each applicant report the effects of the proposed uprate on its core damage frequency and frequencies of large magnitude radioactive release and indicates that this process may be as simple as reporting that the applicant's review of its IPE/PRA found that none of the items previously discussed are changed as a result of the uprate; but it may be as complex as reevaluating the logic model to obtain new dominant cutsets that reflect the significant changes in multiple IPE/PRA assumptions and inputs.

In September 1998, the staff proposed guidelines for the staff's risk review of power uprates (Reference 4). These guidelines, as well as the guidance in Appendix D of SRP Chapter 19, have formed the basis and focus for the current risk reviews of power uprate license applications. The lessons learned from past power uprate reviews have been integrated into the development of this guidance and in establishing the staff's expectations for future reviews of extended power uprate license applications.

This guidance is provided to aid the staff in conducting the risk review of a licensee's application for an extended power uprate, leading up to a determination regarding the potential for the existence of "special circumstances," as defined by Appendix D of Chapter 19 of the SRP. Specific guidance is provided for the scope of the review, the risk information needed to perform the review, the staff review guidance to use in determining the acceptability of the license application and in determining if special circumstances may exist that would warrant invoking the special circumstances notification and review process of Appendix D to SRP Chapter 19, and the review process and documentation requirements for this risk review.

2. SCOPE OF REVIEW

Consistent with SRP Chapter 19 and Regulatory Guide (RG) 1.174 (Reference 5), the licensee's risk analyses used to support a license application and the level of detail of the staff review of those analyses, should be commensurate with the role that the risk results play in the utility's and staff's decisionmaking processes and should be commensurate with the degree of rigor needed to provide a valid technical basis for the staff's decision. As for extended power uprates, the licensees do not request the relaxation of any deterministic requirements for their proposed power uprates and the staff's approval is primarily based on the licensee meeting the current deterministic engineering requirements.

Thus, the purpose of the staff's risk review is to determine if there are any issues that would potentially rebut the presumption of adequate protection provided by the licensee meeting the deterministic requirements and regulations. Such issues could represent the “special circumstances” that would require a more detailed risk review to determine the acceptability of the extended power uprate license application. These reviews can require an extensive level of effort depending upon the required plant modifications to implement the extended power uprate, the plant-specific features and/or vulnerabilities, and the quality of the licensee’s supporting analyses. These reviews need to address the risk impacts to core damage frequency (CDF) and large early release frequency (LERF) due to internal events, external events, and shutdown operations. In addition, these reviews need to address the quality of the licensee’s analyses that are used to support the license application, including addressing any issues or weaknesses that may have been raised in the previous staff reviews of the licensee’s individual plant examinations (IPEs) and individual plant examinations of external events (IPEEE) or by an industry peer review. Further, if the licensee’s results indicate a significant risk impact or if there are significant questions regarding the licensee’s supporting analyses, a site audit of these areas may be deemed appropriate. A site audit might also be performed to resolve PRA quality questions by auditing the licensee’s PRA-related procedures and processes and reviewing their evaluations and resolutions of previous PRA reviews, including the IPE, IPEEE, and industry peer review findings.

If special circumstances are identified, additional information and analyses beyond those identified in this guidance may be required for the staff to be able to determine the acceptability of the license application. This may require the licensee and/or staff to obtain more detailed information to support performing detailed quantitative analyses (e.g., perform seismic PRA instead of reliance on seismic margins analysis or perform shutdown PRA instead of reliance on shutdown outage risk management guidance) to determine the acceptability of the license application. This guidance does not address these review details, which should be mainly focused on the issue(s) creating the circumstances and other considerations as directed by NRC management per the process described in Appendix D of SRP Chapter 19.

3. RISK INFORMATION NEEDED FOR REVIEW

The guidance in this section addresses the information needed by the staff to evaluate the acceptability of the risks and to determine if the potential for special circumstances exist.

3.1 Internal Events Risk Information

The licensee needs to address the risk impacts to the internal events analyses associated with implementing the extended power uprate. Specifically, the licensee needs to address the impacts of the extended power uprate on initiating event modeling and frequencies, component and system reliability and response times, operator response times and associated error probabilities, and functional and system-level success criteria, as well as the overall impact of internal events on CDF and LERF. The discussion of the impacts due to the extended power uprate should include an explanation of why the impacts occur and, where applicable, the quantification of these impacts (e.g., the reduction in operator response timing and revised operator error probabilities).

In addition, if there are any impacts on the PRA results from any other areas that either are affected by the power uprate or are being implemented in parallel with the power uprate (e.g., emergency operating procedure changes, changes in maintenance activities or approach, turbine trip setpoint changes, improved turbine bypass capability, condensate/feedwater modifications or operational changes, main transformer modifications, increased burnup, and longer cycles), then the potential impact of these changes also need to be addressed. For example, if there is a plant modification associated with the uprate that may affect an initiating event (e.g., addition of automatic recirculation system runback on feedwater pump trip), then the initiating event (e.g., loss of feedwater) may need to be explicitly modeled to account for new potential impacts (e.g., spurious runback at full power or failure to runback upon feedwater pump trip). If generic or plant-specific data is used to derive the initiating event frequency, instead of using an explicit model, then the applicability of the data to the new operating conditions will need to be justified. Further, note that the new operating conditions may also impact the top-level, functional plant response (i.e., event tree) modeling. This may then require revising the modeling of and inputs to the best estimate thermal-hydraulic code used to support the development of functional and/or system-level success criteria. The licensee's submittal would also need to describe these modeling, supporting analyses, and success criteria impacts.

The licensee also needs to address the scope, level of detail, and quality of their PRA and other relied upon evaluations (e.g., thermal-hydraulic analyses) used to support their determination that the plant risk is acceptable. The licensee should describe how they ensure that the PRA adequately models the as-built, as-operated plant and that the analyses supporting the extended power uprate adequately reflects how the plant will be operated and configured for the extended power uprate plant conditions. This discussion should specifically address any vulnerabilities, weaknesses, or review findings identified in the IPE, the staff safety evaluation reports or contractor technical evaluation reports on the IPE, and/or any independent/industry peer review findings that could impact the PRA results and conclusions pertinent to this application. The licensee's information needs to be sufficient for the staff to conclude that their PRA and other relied upon evaluations adequately reflect the as-built, as-operated plant for the specific extended power uprate license application.

It is expected that if a peer review has been performed on the PRA that the licensee will present the overall findings of the review (by element) and discuss any elements that were rated low (e.g., less than a 3 on a scale of 1 to 4) and any findings and observations that could potentially impact the licensee's proposed extended power uprate. To address these findings and observations, the licensee may need to perform sensitivity calculations that address the specifically identified weaknesses (e.g., removing credit for equipment repair and recovery). In addition, if the licensee's IPE/PRA took credit for modifications or improvements that had not been implemented, then the licensee needs to explicitly address these conditions. For these areas, the licensee needs to indicate if the improvements have been implemented in accordance with the assumptions and conditions identified in the IPE/PRA. If they have not been implemented, then the licensee needs to provide either a qualitative or quantitative justification for the acceptability of the existing situations for the post-uprate plant conditions.

In addition, some licensees have performed their evaluations of the risk impacts of the extended power uprate prior to having fully determined the plant modifications that will be implemented. In these situations, the licensee needs to justify that their evaluations properly address the potential risk impacts due to the extended power uprate. If there are some modifications that are proposed that may not be implemented (i.e., the final decision of making the modification has not been made or the licensee may wait to see how the equipment performs at uprated power conditions before deciding if a change is needed), then a sensitivity calculation of the risk impacts assuming these modifications are not implemented should be performed. If the design of a modification has not been established at the time of the risk evaluation, then the licensee needs to justify that the assumed design features and resulting failure probabilities bound the proposed modification. Again, a sensitivity calculation may be used to show the impact of different design modifications and/or failure probabilities. If multiple sensitivity calculations are performed to address the above situations, then there should be at least a combination sensitivity calculation performed that combines the adverse impacts of the individual sensitivity calculations.

If the estimated change in CDF and/or LERF, or base CDF and/or LERF, exceeds the RG 1.174 guidelines, including the results of any sensitivity calculations, the licensee should provide a more detailed justification to support the acceptability of implementing the extended power uprate. The licensee's information needs to be sufficient for the staff to conclude that the risk impact from internal events is acceptable and does not create special circumstances.

3.2 External Events Risk Information

The licensee needs to address the risk impacts from external events associated with implementing the extended power uprates. Based on previous reviews, the main issues have involved the analyses and assumptions that date back to the original IPEEE in which credit was taken for plant modifications that had not yet been performed (e.g., taking credit for fixing low-capacity seismic outliers or re-routing cables to eliminate them from certain rooms). Another issue that has been identified is related to the licensee's use of non-PRA type methods in performing their analyses (e.g., margins or vulnerability type analyses). To resolve some of these issues, licensees have had to provide additional information, including performing additional analyses or simplified risk calculations, to show that the risks associated with these outliers or vulnerabilities are acceptable under both current and uprated power conditions. In addition, the staff has performed some simplified calculations, based on the licensee's seismic margins analysis results, to provide a quantitative seismic risk perspective.

If the licensee has a PRA for some external events, the licensee should describe the risk impacts associated with implementing the extended power uprate for these external events and demonstrate that the calculated risk contribution is acceptable. However, if the licensee does not have a PRA for some external events, such as if a margins-type analysis was performed as part of their IPEEE, they should describe how the extended power uprate affects these external events analysis results and conclusions.

The licensee also needs to address the scope, level of detail, and quality of their external events PRA and/or other relied upon evaluations (e.g., seismic margins analysis) used to support their determination that the risk is acceptable. The licensee should describe how they ensure that the analyses adequately represent the as-built, as-operated plant and that the analyses supporting the extended power uprate adequately reflects how the plant will be operated and configured for the extended power uprate plant conditions. Further, if vulnerabilities, outliers, anomalies, or weaknesses were identified in their IPEEE, the associated IPEEE staff safety evaluation reports, IPEEE contractor technical evaluation reports, or industry peer reviews or if the licensee took credit for plant modifications that had not been implemented when the analysis was conducted (e.g., seismic A-46 modifications), the licensee should identify these conditions, how they have resolved these conditions for the extended power uprate, and demonstrate, either quantitatively or qualitatively, that the risk associated with these external events are acceptable. This may involve performing additional analyses or simplified risk calculations that address the specifically identified weaknesses or evaluates the risk implications of the existing conditions (e.g., removing credit for seismic modifications not implemented). The licensee's information needs to be sufficient for the staff to conclude that their external events analyses adequately reflect the as-built, as-operated plant for the specific extended power uprate license application.

If the estimated risk contributions exceed the RG 1.174 guidelines, including the consideration of the existence of a potential vulnerability that is identified in a margins-type analysis or if new potential vulnerabilities are introduced by the extended power uprate, the licensee should provide a more detailed justification to support the acceptability of implementing the extended power uprate. The licensee's information needs to be sufficient for the staff to conclude that the risk from external events is acceptable and does not create special circumstances.

3.3 Shutdown Operations Risk Information

The licensee needs to address the risk impacts on shutdown operations associated with implementing the extended power uprate and describe the plant's shutdown risk management philosophies, processes, and controls that are relied upon to ensure that the risk impacts of the extended power uprate on shutdown operations is not significant. Based on previous reviews, an extended power uprate typically impacts shutdown operations due to the greater decay heat under these conditions, which causes longer times to reach shutdown, longer times before alternative decay heat removal systems can be used, shorter times to boiling, and shorter times for operator responses.

If the licensee has a shutdown PRA, the licensee should describe the risk impacts associated with implementing the extended power uprate and demonstrate that the calculated risk contribution is acceptable. The licensee should specifically address any changes in initiating event frequencies, component reliability, success criteria, and operator actions that are caused by the extended power uprate. However, most licensees do not have a shutdown PRA. If the licensee does not have a shutdown PRA, they should discuss how the extended power uprate affects shutdown risks, how they manage and control these risks, and address any critical or time-limited conditions to demonstrate that these risks are not significant and are properly managed and controlled at the extended power uprate conditions.

The licensee also needs to address the scope, level of detail, and quality of their shutdown PRA and/or other relied upon evaluations (e.g., outage risk management guidance) used to support their determination that the risk impacts associated with extended power uprate are acceptable. The licensee should describe how they ensure that their approach and/or analyses adequately represent the as-built, as-operated plant and that it reflects how the plant will be operated and configured for the extended power uprate plant conditions. The licensee's information needs to be sufficient for the staff to conclude that their analysis of shutdown operations adequately reflects the as-built, as-operated plant for the specific extended power uprate license application.

If the estimated risk contributions exceed the RG 1.174 guidelines, including the consideration of potential vulnerabilities, weaknesses, or limitations in the licensee's shutdown risk management approach or if new potential vulnerabilities are introduced by the extended power uprate, the licensee should provide a more detailed justification to support the acceptability of implementing the extended power uprate. The licensee's information needs to be sufficient for the staff to conclude that the risk impact of the extended power uprate for shutdown operations is acceptable.

4. REVIEW GUIDANCE

Consistent with the current guidance, the appropriate starting point for determining if the potential for special circumstances exists is the acceptance guidelines provided in RG 1.174. This evaluation should address the risks from internal events, external events, and shutdown operations. However, since the review is primarily directed towards determining if adequate protection is challenged, the focus should be primarily on the base risk evaluations (i.e., CDF, LERF, and no potential vulnerabilities identified from a margins-type analysis) as opposed to the change in risk evaluations (i.e., ΔCDF and ΔLERF). While the primary focus is the base risk evaluation, it is still important to assess the change in risk to understand the magnitude of the risk increase associated with the extended power uprate. Large base risk values or large changes in risk values that surpass the RG 1.174 acceptance guidelines should warrant additional staff scrutiny of the analyses, results, and quality of the licensee's analyses. This would be a factor in determining the need to conduct a site audit of the licensee's PRA and/or their PRA management procedures and processes. If the staff determines that the base risk values are significantly beyond the RG 1.174 acceptance guidelines, then this should invoke the special circumstances process of Appendix D of SRP Chapter 19.

To determine that the analyses used in support of the license application is of sufficient quality, scope, and level of detail, the staff should evaluate the information provided by the licensee using the guidance provided in RG 1.174, as well as consider the staff's previous reviews on the licensee's IPE and IPEEE submittals and the conclusions and findings of any industry or independent peer reviews. The staff needs to be assured that the relied upon analyses adequately reflects the as-built, as-operated plant.

All licensees have at least a Level I internal events PRA, but most licensees do not have a fully integrated PRA that addresses internal events, external events, and shutdown operations. Further, the analyses that are performed for many external events and shutdown operations either are not quantitative in nature or are screening/vulnerability-type analyses that are not performed to the same level of depth and rigor as the internal events analyses. Therefore, the staff may need to rely on some general figures of merit or simplistic calculations to provide a more comprehensive perspective of the potential risks associated with a licensee's extended power uprate application.

For example, in addressing the risk impacts for shutdown operations in the absence of a licensee's shutdown PRA, the review staff should refer to SECY 97-168, "Issuance for Public Comment of Proposed Rulemaking Package for Shutdown and Fuel Storage Pool Operation," in which the staff provides estimates of shutdown risk for various interpretations of the industry guidance. The risk estimates cited in SECY 97-168 were not meant to bound plant operations, but were intended to be examples of reasonable interpretations of industry guidance. Depending on the specific licensee's approach to managing shutdown risks, an estimate of the magnitude of the risk for shutdown operations can be determined using SECY 97-168. An example of this review approach is provided as Attachment 3 to Matrix 13 of RS-001.

As a further example, in addressing the risk impacts related to seismic events for situations in which the licensee has performed a seismic margins analysis instead of a seismic PRA, the review staff may need to perform a simplistic calculation to determine the magnitude of the seismic risk. An approximation method is provided in a paper by Robert P. Kennedy entitled *"Overview of Methods for Seismic PRA and Margin Analysis Including Recent Innovations,"* (Reference 6) that uses the plant's high confidence of a low probability of failure (HCLPF) value that is determined by the licensee's seismic margins analysis and the site's seismic hazard curve that is based on NUREG-1488 (Reference 7) to derive an approximation of the magnitude of the risk associated with seismic events. An example of this calculation is provided as Attachment 4 to Matrix 13 of RS-001.

The results of these simplistic approaches should not be used as the sole basis for determining the acceptability or rejection of a license application, but rather should be used to gain perspective into the risks associated with these events/operations, insights into the licensee's management of these risks, and a focus for areas that may require further review or may indicate the potential for special circumstances. If these results indicate the potential for significantly exceeding the RG 1.174 acceptance guidelines (i.e., indicating the potential existence of special circumstances), then the staff should pursue these risk aspects further with the licensee and seek more information and analyses to more accurately define these risk contributors. If the licensee cannot or will not be able to provide the additional information or analyses in a timely fashion, then the staff should progress in its review of the risk information and notify management of this potential for special circumstances.

If issues are identified that could rebut the presumption of adequate protection (i.e., special circumstances), the process delineated in Appendix D of Chapter 19 of the SRP should be implemented. This process is also described in Regulatory Issue Summary (RIS) 2001-02, "Guidance on Risk-Informed Decisionmaking in License Amendment Reviews," and includes informing/engaging the licensee and NRC management regarding the risk concern, obtaining management approval to request additional risk information, and to evaluate this risk information to determine if there is reasonable assurance of adequate protection. If the NRC management agrees with the staff that a special circumstance appears to exist, there is also direction to notify the Commission of this decision. The rationale that led to the expansion of the depth of the review, as well as the findings of the associated review, should be documented in the staff's safety evaluation.

5. RISK REVIEW PROCESS AND DOCUMENTATION

The SPSB Safety Program Section staff should document their review activities associated with extended power uprate license applications through the issuance of a safety evaluation, which, upon management approval, is subsequently transmitted to the responsible project manager to incorporate into the NRC safety evaluation report on the license application. The review activities leading up to the development of the staff safety evaluation are described in this section.

In initiating the risk review, the staff should first perform an "acceptance review" of the information provided by the licensee. The acceptance review should ensure that the licensee's submittal meets the intent of Section 3 of this guidance. The information provided by the licensee needs to be sufficient for the staff to be able to make a determination regarding special circumstances, based on the guidance described in Section 4. If the licensee's information, provided in accordance with Section 3 of this guidance, combined with any staff independent and/or simplified calculations, performed in accordance with Section 4 of this guidance, indicates that the overall plant risks are well below the acceptance guidelines of RG 1.174 and that there are no special circumstances, the staff may not develop a detailed safety evaluation. Instead, the staff may provide an abbreviated safety evaluation that documents that the licensee's submittal, combined with any staff independent and/or simplified calculations, has adequately addressed the risks associated with the extended power uprate and that these risks have been shown to be acceptably small.

If the staff identifies any issues with the licensee's submittal or needs to clarify any information provided by the licensee, then the staff should pursue these areas initially through the issuance of requests for additional information (RAIs). Some issues, such as a lack of information about expected risk contributors or differences between the supporting analyses and the actual plant operations, may be resolved through RAIs or by conducting a site audit of the licensee's pertinent documentation and/or processes, without needing to invoke the process for special circumstances. If issues are identified that could indicate the potential for special circumstances, then these issues should be elevated to management as early as possible during the staff review since such a determination may invoke a detailed review process and mean that the project schedules and staff-hour estimates will need to be revised.

Through the staff reviews, a number of issues may be identified with specific aspects of the risk analyses used to support a licensee's application for an extended power uprate. The main issues that have been identified have involved the change in risk calculation when bounding or conservative values are used in the base risk model and the reliance on external events analyses and assumptions that date back to the original IPEEE (e.g., taking credit for fixing low-capacity seismic outliers or re-routing cables to eliminate them from certain rooms). In some of these cases, the licensee has had to provide additional information, including performing additional analyses or simplified calculations, to make the relied upon analyses more reflective of the actual plant conditions and to show that the associated risks are acceptable under both current and uprated power conditions. However, being a non-risk-informed submittal review, the staff focus is primarily on determining if there are any conditions associated with implementing the extended power uprate that would significantly alter the current practices of the licensees or create new vulnerabilities, such that issues are raised that could rebut the presumption of adequate protection provided by meeting the deterministic requirements and regulations. If these circumstances arise, the staff should seek to perform a more in-depth review to determine the appropriateness of accepting the extended power uprate license application or if there would be grounds warranting denial of the licensee's application for an extended power uprate. However, if the identified issues do not raise adequate protection questions, the issues should be documented in the safety evaluation and clearly explained as why they do not rise to this level of concern.

The staff safety evaluation should address the staff's findings and conclusions for each of the major review areas (i.e., internal events, external events, and shutdown operations), including the quality of the licensee's analyses supporting these areas (i.e, PRA, margins-type analyses, vulnerability assessments, etc.), and if any issues were identified that could potentially create special circumstances. The results of any detailed review required by a determination of special circumstances should also be documented in the safety evaluation. In performing the review, the staff may also identify issues related to the licensee's supporting analyses that do not affect the determination regarding special circumstances for the extended power uprate license application. These issues should be identified within the staff safety evaluation, with an explanation as to why they do not impact the extended power uprate license application.

In addition to the primary task of performing the risk review, the Safety Program Section staff may be requested by other NRC technical review branches to provide risk analyses and/or insights to support the evaluations of potential impacts that are identified in these other branches' review areas. The results associated with these requested evaluations should be integrated directly within the safety evaluations of the technical branch(es) that requested the support. Thus, there should not be a separate input from the SPSB Safety Program Section in these requested support areas, unless it impacts the staff risk review findings.

6. REFERENCES

1. U.S. Nuclear Regulatory Commission, *Use of Probabilistic Risk Assessment in Plant-Specific, Risk-Informed Decisionmaking: General Guidance*, NUREG-0800, Standard Review Plan Chapter 19.0, Revision 1, December 2002.

2. U.S. Nuclear Regulatory Commission, *Guidance on Risk-Informed Decisionmaking in License Amendment Reviews*, Regulatory Issue Summary 2001-02, January 18, 2001.

3. Letter from Dennis M. Crutchfield (NRC) to G. L. Sozzi (GENE), *Staff Position Concerning General Electric Boiling Water Reactor Extended Power Uprate Program*, February 8, 1996.

4. Memorandum from Richard J. Barrett (NRC) to Gary M. Holahan (NRC), *Proposal for a Guideline on Risk-Informed Staff Review of Power Uprate*, September 22, 1998.

5. U.S. Nuclear Regulatory Commission, *An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis*, Regulatory Guide 1.174, July 1998.

6. Kennedy, R.P., *Overview of Methods for Seismic PRA and Margin Analysis Including Recent Innovations*, Proceedings of the OECD-NEA Workshop on Seismic Risk, Tokyo, Japan, August 1999.

7. U.S. Nuclear Regulatory Commission, *Revised Livermore Seismic Hazard Estimates for Sixty-Nine Nuclear Power Plant Sites East of the Rocky Mountains*, NUREG-1488, April 1994.

ATTACHMENT 3 TO MATRIX 13

Example Staff Review of Shutdown Risk Based on SECY 97-168

Risk Evaluation

In SECY 97-168, "Issuance for Public Comment of Proposed Rulemaking Package for Shutdown and Fuel Storage Pool Operation," the staff provided two estimates of pressurized water reactor (PWR) shutdown risk, which credited equipment required by technical specification (TS) and equipment recommended to be available based on guidance from generic letter (GL) 88-17 and NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management." These two "voluntary action cases" represent different interpretations of NUMARC 91-06 and GL 88-17. These two cases were not meant to bound plant operations, but were intended to be examples of reasonable interpretations of industry guidance. These two cases cover cold shutdown operations and refueling operations until the refueling cavity is flooded. Reduced inventory operations are a subset of this condition.

The high core damage frequency (CDF) voluntary action case represents a minimal level of implementation of both guidance documents in terms of the amount of extra equipment and additional sources of water being made available. For PWRs, the higher CDF voluntary action case includes the equipment credited by TS, based on Westinghouse standard TS, plus one emergency core cooling system (ECCS) pump, gravity feed, and an "available" containment. An "available" containment is defined as one that can be closed by remote or local manual actions before containment conditions become intolerable. The high case had a CDF estimate of 8E-5/year.

The low CDF voluntary action case represents a more in-depth implementation of both guidance documents. The lower CDF case adds an additional emergency diesel generator (EDG) or equivalent power source, a second ECCS pump, containment spray pumps to supplement the residual heat removal (RHR) pumps, and an enhanced recirculation capability. The low case had a CDF estimate of 2E-6/year

Based on the licensee's shutdown cooling control procedures, the operators should have an high pressure safety injection (HPSI) flow path available at all times unless the reactor vessel is defueled. During reduced inventory operations, the licensee maintains a second flow path in addition to the HPSI flow path. However, based on conversations with the licensee, the second flow path may be a small charging pump that may not have the capability to keep the core covered following a loss of inventory event that includes a loss of both the RHR flow path, which is the normal means of decay heat removal, and the HPSI flow path.

Concerning the licensee's containment closure capability, the outage risk management guidelines (ORMGs) allow for a containment breach that cannot be closed prior to the estimated time to boiling. However, the licensee maintains that such a breach would not be incorporated into the outage schedule and, based on discussions with the licensee, such breaches would be unanticipated and/or inadvertent. The small increase in decay heat due to the proposed extended power uprate (EPU) will reduce the time available for operator actions, such as to achieve containment closure. However, even for the most time-limiting closure action (i.e., the equipment hatch), which the licensee has demonstrated a closure capability of within 5 minutes to 15 minutes, the estimated time to boiling would be greater than 18 minutes for EPU conditions as opposed to over 20 minutes for the pre-EPU conditions. Therefore, the operator's ability to inject before core damage and the ability to close containment before boiling should not be significantly changed, since (1) there is margin between the time-limiting actions and the time to boiling, (2) the operators regularly calculate the time to boiling, and (3) the licensee maintains the availability of the core exit thermocouples to monitor reactor coolant system (RCS) temperature until preparations for vessel head removal.

Based on the staff's review of the licensee's shutdown mitigation capability provided by the licensee's responses to the staff's requests for additional information, the licensee's shutdown mitigation capability appears to be closer to the high CDF voluntary action case.

ATTACHMENT 4 TO MATRIX 13

Example Staff Review of Seismic Risk Using Simplified Calculations

Risk Evaluation

The safety evaluation report (SER) on the licensee's individual plant examination of external events (IPEEE) indicated, based on the staff's screening review, that the licensee's process is capable of identifying the most likely severe accidents and severe accident vulnerabilities and therefore, that the licensee had met the intent of Supplement 4 to generic letter (GL) 88-20. For the IPEEE seismic analysis, the licensee's plant is categorized as a 0.3g focused-scope plant, per NUREG-1407. The licensee performed the seismic evaluation using the Electric Power Research Institute (EPRI) seismic margins analysis (SMA) methodology, as described in EPRI NP-6041-SL.

Because the licensee used the EPRI SMA methodology, they did not quantify a seismic core damage frequency (CDF). However, the licensee states in their supplemental information for the extended power uprate (EPU) license amendment that the conclusions and results of the SMA were judged to be unaffected by the EPU. Further, they state that the EPU has no impact on the seismic qualifications of the systems, structures, and components. Specifically, the EPU results in additional thermal energy stored in the reactor pressure vessel (RPV), but the additional blowdown loads on the RPV and containment given a coincident seismic event are judged not to alter the results of the SMA.

The SER on the IPEEE indicates that the licensee had implemented a number of improvements during the resolution of unreviewed safety issue (USI) A-46 and that a number of additional improvements were still under consideration. The licensee indicated that any necessary design changes to address these items would be completed in conjunction with the approved schedule for resolution of the USI A-46 outliers. In particular, the SER states that the licensee was developing a concept for providing a seismically-qualified/verified make-up path for a particular accident scenario. The licensee's IPEEE SMA took credit for this plant modification and related operational changes needed to implement the seismically-qualified/verified make-up feature. However, these plant modifications had not been implemented at the time of the original EPU license amendment submittal. Thus, it appears that the IPEEE SMA does not accurately represent the as-built, as-operated plant. Therefore, the staff requested that the licensee augment their IPEEE SMA by performing some simplified seismic risk evaluations of the current and EPU plant configurations for the outlier scenario (i.e., non-seismically qualified make-up source). In addition, the staff performed an independent simplistic calculation to estimate the magnitude of the seismic risk associated with the identified outlier condition.

For this scenario, though the IPEEE indicates that it is a 0.3g focused-scope SMA, the scenario involves equipment with an high confidence of a low probability of failure (HCLPF) value that is much lower than 0.3g. The scenario involves a seismic event that involves the failure of the non-seismically-qualified makeup source, which has a HCLPF value of 0.15g peak ground acceleration (PGA). The licensee's results indicate that the current, pre-uprate plant and the

EPU plant CDF values for this scenario are both about 1E-5/year, with a change in risk due to the uprate of about 1E-8/year.

The staff used the approximation method provided in a paper by Robert P. Kennedy entitled *"Overview of Methods for Seismic PRA and Margin Analysis Including Recent Innovations."* This approach uses the plant's HCLPF value that is determined by the licensee's SMA and the site's seismic hazard curve that is based on NUREG-1488 to derive an approximation of the magnitude of the risk associated with seismic events. The staff's independent simplistic calculation used a plant HCLPF value of 0.15g PGA, since that is the HCLPF of the non-seismically-qualified makeup source, and the recommended logarithmic standard deviation of 0.4. Using these values, the seismic CDF for the outlier scenario is estimated to be approximately 1.7E-5/year. The seismic risk associated with the remainder of the plant having a HCLPF at 0.3g PGA using the same approach is about 3.1E-6/year. Thus, based on the staff's approximation, the total seismic CDF is estimated to be about 2E-5/year.

SECTION 3

DOCUMENTATION OF REVIEW

3.1 Documenting Reviews of Extended Power Uprate Applications

This section includes two template safety evaluations for use in generating plant-specific safety evaluations: one for boiling-water reactor (BWR) plants and one for pressurized-water reactor (PWR) plants. These template safety evaluations were developed consistent with NRR Office Instruction LIC-101, "License Amendment Review Procedures."

When preparing plant-specific safety evaluations, Project Managers have the lead for completing Sections 1.0, 3.0, 4.0, 6.0, 7.0, 8.0, and 9.0 of the template safety evaluation. Reviewers with primary review responsibility identified in the matrices in Section 2.1 of this review standard have the lead for completing the subsections of Section 2.0 of the template safety evaluations that correspond to the areas within their branch's primary review responsibility. Reviewers with primary review responsibility also have the lead for completing Section 5.0 of the template safety evaluation. Project Managers are responsible for preparing and finalizing the plant-specific safety evaluation, including consolidating the inputs received from other branches.

When preparing plant-specific safety evaluations, follow the instructions below.

(1) Use the applicable template safety evaluation in Section 3.2 (for BWRs) or Section 3.3 (for PWRs) of this review standard.

(2) Replace the information within the brackets with applicable plant-specific information.

(3) Based on the results of the technical review performed in accordance with Section 2.1 of this review standard, for each technical area of the template safety evaluation where the licensing basis of the plant has been identified as different from the guidance provided in the documents referenced in the "SRP Section Number" and "Other Guidance" columns of the matrices, modify the "Regulatory Evaluation" and "Conclusion" sections to be consistent with the licensing basis of the plant. Ensure that the changes are written consistent with the format and content of the template safety evaluation.

(4) Based on the results of the technical review performed in accordance with Section 2.1 of this review standard, if additional technical areas beyond those identified in the matrices in Section 2.1 of this review standard are necessary, address the additional technical areas under the "Additional Review Areas" subsection of the appropriate section of the safety evaluation. Provide a regulatory evaluation, technical evaluation, and conclusion for each of the additional technical areas. Ensure the additional sections are written consistent with the format and content of the template safety evaluation.

(5) Based on the results of the technical review performed in accordance with Section 2.1 of this review standard, if a technical area is determined to not be applicable or necessary for the plant under review, keep that section's heading in the safety evaluation, delete the "Regulatory Evaluation" and "Conclusion" sections for that area, and discuss the reasons why a review of that particular technical area is not needed.

(6) Summarize the technical review and findings in the appropriate "Technical Evaluation" section of the safety evaluation.

(7) Discuss independent calculations performed to support the review in the appropriate "Technical Evaluation" section of the safety evaluation.

(8) Review the "Conclusion" sections of the safety evaluation and modify them, as necessary, to reflect the conclusions reached as a result of the review.

(9) Identify areas for consideration by the NRC's inspection staff in the "Recommended Areas for Inspection" section of the safety evaluation. Each area identified should include a rationale. The identified areas are not intended to be inspection requirements, but are provided to give the inspectors insight into important bases for approving the EPU.

(10) Generate a detailed table of contents for the final plant-specific safety evaluation. The detailed table of contents should include a listing of all areas addressed within each insert.

SECTION 3.2 of RS-001

TEMPLATE SAFETY EVALUATION

for

BOILING-WATER REACTOR EXTENDED POWER UPRATE

TABLE OF CONTENTS

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. ____ TO FACILITY OPERATING LICENSE NO. **[XXX-XX]**

[NAME OF LICENSEE]

[NAME OF FACILITY]

DOCKET NO. 50-**[XXX]**

1.0 INTRODUCTION

1.1 Application

By application dated [ ], as supplemented by letter**[s]** dated [ ], the **[Name of Licensee]** (the licensee) requested changes to the Facility Operating License and Technical Specifications (TSs) for the **[Plant Name]**. The supplemental letter[s] dated [ ], provided additional clarifying information that did not expand the scope of the initial application and did not change the Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on **[date] (XX FR XXXX).**

The proposed changes would increase the maximum steady-state reactor core power level from **[current licensed power level]** megawatts thermal (MWt) to **[power level proposed by the licensee]** MWt, which is an increase of approximately **[##]** percent. The proposed increase in power level is considered an extended power uprate (EPU).

1.2 Background

[Plant Name] is a boiling-water reactor (BWR) plant of the BWR/**[#]** design with a Mark-**[#]** containment. **[Plant Name]** has the following special features/unique designs:

[Insert any special features/unique designs]

The NRC originally licensed **[Plant Name]** on **[date]** for operation at **[original licensed power level]** MWt. **[By Amendment No. [###] dated [ ], the NRC granted a power uprate to [Plant Name] of [##] percent, allowing the plant to be operated at [current licensed power level] MWt.] Therefore, the proposed EPU would result in an increase of approximately [##] percent over the original licensed power level [and [##] percent over the current licensed power level] for [Plant Name].]**

1.3 Licensee's Approach

The licensee's application for the proposed EPU follows the guidance in the Office of Nuclear Reactor Regulation's (NRR's) Review Standard (RS)-001, "Review Standard for Extended Power Uprates," to the extent that the review standard is consistent with the licensing basis of the plant. Where differences exist between the plant-specific licensing basis and RS-001, the licensee described the differences and provided evaluations consistent with the licensing basis of the plant. The licensee also used **[Identify topical reports or other documents used by the licensee for guidance related to the scope of the proposed EPU; NRC staff approvals, ranges of applicability, any limitations/restrictions associated with the documents; and consistency of the licensee's application with the ranges of applicability and limitations/restrictions. The discussion in this section is to cover topical reports and other documents referenced for the overall power uprate process. It is not intended to cover topical reports and other documents for specific methods of analyses. Topical reports and other documents referenced for specific methods of analyses are to be covered in the applicable technical evaluation section of this safety evaluation].**

Insert this sentence if the licensee is planning to implement the EPU in one stage.
[The licensee plans to implement the EPU in one step. The licensee plans to make the modifications necessary to implement the EPU during the refueling outage in [season year (e.g., fall 2003)]. Subsequently, the plant will be operated at [##] MWt starting in Cycle [##].]

Insert this paragraph if the licensee is planning to implement the EPU in stages:
[The licensee plans to implement the EPU in [#] steps of [## and ##] percent. The licensee plans to make modifications necessary to implement the first step during the refueling outage in [season year (e.g., fall 2003)]. Subsequently, the plant will be operated at [##] MWt during Cycle [##]. The remainder of the modifications will be completed during the refueling outage in [season year (e.g., fall 2003)], with subsequent operation at [##] MWt starting in Cycle [##].]

1.4 Plant Modifications

The licensee has determined that several plant modifications are required to achieved the proposed EPU. The following is a list of these modifications and the licensee's proposed schedule for completing them.

[Provide a list of plant modifications.]

The NRC staff's evaluation of the licensee's proposed plant modifications is provided in Section 2.0 of this safety evaluation.

1.5 Method of NRC Staff Review

The NRC staff reviewed the licensee's application to ensure that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) activities proposed will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public. The purpose of the NRC staff's review is to evaluate the licensee's assessment of the impact of the proposed EPU on licensing-basis analyses. The NRC staff evaluated the licensee's application and supplements. The NRC staff also evaluated **[Include additional review items, as necessary (e.g., audits of certain information at the plant and vendor sites, and independent analyses), for areas where such analyses were deemed appropriate by the NRC staff]**.

In areas where the licensee and its contractors used NRC-approved or widely accepted methods in performing analyses related to the proposed EPU, the NRC staff reviewed relevant material to ensure that the licensee/contractor used the methods consistent with the limitations and restrictions placed on the methods. In addition, the NRC staff considered changes in plant operating conditions on the use of these methods to ensure that the methods are appropriate for use at the proposed EPU conditions. Details of the NRC staff's review are provided in Section 2.0 of this safety evaluation.

Audits of analyses supporting the EPU were conducted in relation to the following topics:

[Provide a list of areas for which audits were performed.]

The results of the audits are discussed in section 2.0 of this safety evaluation.

Independent NRC staff calculations were performed in relation to the following topics:

[Provide a list of areas for which independent NRC staff calculations were performed.]

The results of the calculations are discussed in section 2.0 of this safety evaluation.

2.0 EVALUATION

2.1 Materials and Chemical Engineering

SEE INSERT 1 FOR SECTION 3.2 OF RS-001

2.2 Mechanical and Civil Engineering

SEE INSERT 2 FOR SECTION 3.2 OF RS-001

2.3 Electrical Engineering

SEE INSERT 3 FOR SECTION 3.2 OF RS-001

2.4 Instrumentation and Controls

SEE INSERT 4 FOR SECTION 3.2 OF RS-001

2.5 Plant Systems

SEE INSERT 5 FOR SECTION 3.2 OF RS-001

2.6 Containment Review Considerations

SEE INSERT 6 FOR SECTION 3.2 OF RS-001

2.7 Habitability, Filtration, and Ventilation

SEE INSERT 7 FOR SECTION 3.2 OF RS-001

2.8 Reactor Systems

SEE INSERT 8 FOR SECTION 3.2 OF RS-001

2.9 Source Terms and Radiological Consequences Analyses

SEE INSERT 9 FOR SECTION 3.2 OF RS-001

2.10 Health Physics

SEE INSERT 10 FOR SECTION 3.2 OF RS-001

2.11 Human Performance

SEE INSERT 11 FOR SECTION 3.2 OF RS-001

2.12 Power Ascension and Testing Plan

SEE INSERT 12 FOR SECTION 3.2 OF RS-001

2.13 Risk Evaluation

SEE INSERT 13 FOR SECTION 3.2 OF RS-001

3.0 FACILITY OPERATING LICENSE AND TECHNICAL SPECIFICATION CHANGES

To achieve the EPU, the licensee proposed the following changes to the Facility Operating License and TSs for **[Plant Name]**.

[Provide a list of license and TSs changes (including license conditions) and an NRC staff evaluation of each.]

4.0 REGULATORY COMMITMENTS

Insert the following sentence if the licensee has not made any regulatory commitments in support of the EPU.
The licensee has made no regulatory commitments in its application for the EPU.

Insert the following if the licensee has made regulatory commitments in support of the EPU.
The licensee has made the following regulatory commitment(s):

[Provide a summary of each regulatory commitment made by the licensee.]

The NRC staff finds that reasonable controls for the implementation and for subsequent evaluation of proposed changes pertaining to the above regulatory commitment(s) are best provided by the licensee's administrative processes, including its commitment management program. The above regulatory commitments do not warrant the creation of regulatory requirements (items requiring prior NRC approval of subsequent changes).

5.0 RECOMMENDED AREAS FOR INSPECTION

The NRC staff has conducted an extensive review of the licensee's plans and analyses related to the proposed EPU and concluded that they are acceptable. The NRC staff review has identified the following areas for consideration by the NRC inspection staff during the licensee's implementation of the proposed EPU. These areas are recommended based on past experience with EPUs, the extent and unique nature of modifications required to achieve the proposed EPU, and new conditions of operation required for the proposed EPU. They do not constitute inspection requirements, but are intended to give inspectors insight into important bases for approving the EPU.

[Provide list of recommended areas for inspection.]

6.0 STATE CONSULTATION

In accordance with the Commission's regulations, the **[Name of State]** State official was notified of the proposed issuance of the amendment. The State official had **[no]** comments. **[If comments were received, address them here]**

7.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, 51.33, and 51.35, a draft Environmental Assessment and finding of no significant impact was prepared and published in the *Federal Register* on **[Date (## FR #####)]**. The draft Environmental Assessment provided a 30-day opportunity for public comment. **[No]** comments were received on the draft Environmental Assessment. **[If comments were received, address them here.]** The final Environmental Assessment was published in the *Federal Register* on **[Date (## FR #####)]**. Accordingly, based upon the environmental assessment, the Commission has determined that the issuance of this amendment will not have a significant effect on the quality of the human environment.

8.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

9.0 REFERENCES

1. RS-001, "Review Standard for Extended Power Uprates," Date.

2. **[Insert additional references as necessary]**

Attachment: List of Acronyms

Principal Contributors:

Date:

LIST OF ACRONYMS

AAC	alternate ac sources
ac	alternating current
ALARA	as low as reasonably achievable
ARAVS	auxiliary and radwaste area ventilation system
ARI	alternate rod insertion
ASME	American Society of Mechanical Engineers
ATWS	anticipated transient without scram
B&PV	boiler and pressure vessel
BL	bulletin
BOP	balance-of-plant
BTP	branch technical position
BWR	boiling-water reactor
BWRVIP	Boiling Water Reactor Vessel and Internals Project
CDF	core damage frequency
CFR	*Code of Federal Reguations*
CFS	condensate and feedwater system
CRAVS	control room area ventilation system
CRDA	control rod drop accident
CRDM	control rod drive mechanism
CRDS	control rod drive system
CUF	cumulative usage factor
CWS	circulating water system
DBA	design-basis accident
DBLOCA	design-basis loss-of-coolant accident
dc	direct current
DG	draft guide
EAB	exclusion area boundary
ECCS	emergency core cooling system

EFDS	equipment and floor drainage system
EPG	emergency procedure guideline
EPRI	Electric Power Research Institute
EPU	extended power uprate
EQ	environmental qualification
ESF	engineered safety feature
ESFAS	engineered safety feature actuation system
ESFVS	engineered safety feature ventilation system
FAC	flow-accelerated corrosion
FHA	fuel handling accident
FPP	fire protection program
GDC	general design criterion
GL	generic letter
I&C	instrumentation and controls
IN	information notice
IPE	individual plant examination
IPEEE	individual plant examination of external events
LERF	large early release frequency
LLHS	light load handling system
LOCA	loss-of-coolant accident
LOOP	loss of offsite power
LPZ	low population zone
MC	main condenser
MCES	main condenser evacuation system
MOV	motor-operated valve
MSIV	main steam isolation valve
MSIVLCS	main steam isolation valve leakage control system
MSLB	main steamline break

MSSS	main steam supply system
MWt	megawatts thermal
NEI	Nuclear Energy Institute
NPSH	net positive suction head
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
NSSS	nuclear steam supply system
O&M	operations and maintenance
P-T	pressure-temperature
PWSCC	primary water stress-corrosion cracking
RCIC	reactor core isolation cooling
RCPB	reactor coolant pressure boundary
RCS	reactor coolant system
RG	regulatory guide
RHR	residual heat removal
RS	review standard
RWCS	reactor water cleanup system
SAFDL	specified acceptable fuel design limit
SAG	severe accident guideline
SAR	Safety Analysis Report
SBO	station blackout
SFP	spent fuel pool
SFPAVS	spent fuel pool area ventilation system
SGTS	standby gas treatment system
SLCS	standby liquid control system
SRP	Standard Review Plan
SSCs	structures, systems, and components
SSE	safe-shutdown earthquake

SWMS	solid waste management system
SWS	service water system
TAVS	turbine area ventilation system
TBS	turbine bypass system
TCV	turbine control valve
TEDE	total effective dose equivalent
TS	technical specification
UHS	ultimate heat sink

INSERT 1

FOR

SECTION 3.2 - BWR TEMPLATE SAFETY EVALUATION

2.1 Materials and Chemical Engineering

2.1.1 Reactor Vessel Material Surveillance Program

Regulatory Evaluation

The reactor vessel material surveillance program provides a means for determining and monitoring the fracture toughness of the reactor vessel beltline materials to support analyses for ensuring the structural integrity of the ferritic components of the reactor vessel. The NRC staff's review primarily focuses on the effects of the proposed EPU on the licensee's reactor vessel surveillance capsule withdrawal schedule. The NRC's acceptance criteria are based on (1) General Design Criterion (GDC)-14 for assuring an extremely low probability of rapidly propagating fractures of the reactor coolant pressure boundary (RCPB); (2) GDC-31 for assuring that the RCPB will behave in a nonbrittle manner and the probability of a rapidly propagating fracture is minimized; (3) 10 CFR Part 50, Appendix H, for determination and monitoring of fracture toughness; and (4) 10 CFR 50.60 for compliance with the requirements of 10 CFR Part 50, Appendix H. Specific review criteria are contained in Standard Review Plan (SRP) Section 5.3.1 and other guidance provided in Matrix 1 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed EPU on the reactor vessel surveillance withdrawal schedule and concludes that the licensee has adequately addressed changes in neutron fluence and their effects on the schedule. The NRC staff further concludes that the reactor vessel capsule withdrawal schedule is appropriate to ensure that the material surveillance program will continue to meet the requirements of GDC-14, GDC-31, 10 CFR Part 50, Appendix H, and 10 CFR 50.60 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the reactor vessel material surveillance program.

2.1.2 Pressure-Temperature Limits and Upper-Shelf Energy

Regulatory Evaluation

Pressure-temperature (P-T) limits are established to ensure the structural integrity of the ferritic components of the RCPB during any condition of normal operation, including anticipated operational occurrences and hydrostatic tests. The NRC staff's review of P-T limits covers the P-T limits' methodology and the calculations for the specified effective full power years, considering neutron embrittlement effects and using linear elastic fracture mechanics. The NRC's acceptance criteria for P-T limits are based on (1) GDC-14 for assuring an extremely low probability of abnormal leakage, rapidly propagating failure, and gross rupture of the RCPB; (2) GDC-31 for assuring that the RCPB will behave in a nonbrittle manner and the probability of a rapidly propagating fracture is minimized; (3) 10 CFR Part 50, Appendix G, for material testing and fracture toughness; and (4) 10 CFR 50.60 for compliance with the requirements of 10 CFR Part 50, Appendix G. Specific review criteria are contained in SRP Section 5.3.2 and other guidance provided in Matrix 1 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed EPU on the P-T limits for the plant and concludes that the licensee has adequately addressed changes in neutron fluence and their effects on the P-T limits. The NRC staff further concludes that the licensee has demonstrated the validity of the proposed P-T limits for the proposed EPU operation. Based on this, the NRC staff concludes that the proposed P-T limits will continue to meet the requirements of GDC-14, GDC-31, 10 CFR Part 50, Appendix G, and 10 CFR 50.60 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the proposed P-T limits.

2.1.3 Reactor Internal and Core Support Materials

Regulatory Evaluation

The reactor internals and core supports include structures, systems, and components (SSCs) that perform safety functions and/or whose failure could affect safety functions performed by other SSCs. These safety functions include reactivity monitoring and control, core cooling, and fission product confinement (within both the fuel cladding and the reactor coolant system (RCS)). The NRC staff's review covers the materials' specifications and mechanical properties, welds, weld controls, nondestructive examination procedures, corrosion resistance, and susceptibility to degradation. The NRC's acceptance criteria for reactor internal and core support materials are based on GDC-1 and 10 CFR 50.55a for material specifications, controls on welding, and inspection of reactor internals and core supports. Specific review criteria are contained in SRP Section 4.5.2 and Boiling Water Reactor Vessel and Internals Project (BWRVIP)-26.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed EPU on the susceptibility of reactor internal and core support materials to known degradation mechanisms and concludes that the licensee has identified appropriate degradation management programs to address the effects of changes in operating temperature and neutron fluence on the integrity of reactor internal and core support materials. The NRC staff further concludes that the licensee has demonstrated that the reactor internal and core support materials will continue to be acceptable and will continue to meet the requirements of GDC-1 and 10 CFR 50.55a following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to reactor internal and core support materials.

2.1.4 Reactor Coolant Pressure Boundary Materials

Regulatory Evaluation

The RCPB defines the boundary of systems and components containing the high pressure fluids produced in the reactor. The NRC staff's review of RCPB materials covers their specifications, compatibility with the reactor coolant, fabrication and processing, susceptibility to degradation, and degradation management programs. The NRC's acceptance criteria for RCPB materials are based on (1) GDC-1 and 10 CFR 50.55a for quality standards; (2) GDC-4 for compatibility of components with environmental conditions; (3) GDC-14 and GDC-31 for assuring an extremely low probability of rapidly propagating fracture or gross rupture of the RCPB; and (4) 10 CFR Part 50, Appendix G, for materials testing and acceptance criteria for fracture toughness of the RCPB. Specific review criteria are contained in SRP Section 5.2.3 and other guidance provided in Matrix 1 of RS-001. Additional review guidance for primary water stress-corrosion cracking (PWSCC) of dissimilar metal welds and associated inspection programs is contained in Generic Letter (GL) 97-01, Information Notice (IN) 00-17, Bulletin (BL) 01-01, BL 02-01, and BL 02-02. Additional review guidance for thermal embrittlement of cast austenitic stainless steel components is contained in a letter from C. Grimes, NRC, to D. Walters, Nuclear Energy Institute (NEI), dated May 19, 2000.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed EPU on the susceptibility of RCPB materials to known degradation mechanisms and concludes that the licensee has identified appropriate degradation management programs to address the effects of changes in system operating temperature on the integrity of RCPB materials. The NRC staff further concludes that the licensee has demonstrated that the RCPB materials will continue to be acceptable following implementation of the proposed EPU and will continue to meet the requirements of GDC-1, GDC-4, GDC-14, GDC-31, 10 CFR Part 50, Appendix G, and 10 CFR 50.55a. Therefore, the NRC staff finds the proposed EPU acceptable with respect to RCPB materials.

2.1.5 Protective Coating Systems (Paints) - Organic Materials

Regulatory Evaluation

Protective coating systems (paints) provide a means for protecting the surfaces of facilities and equipment from corrosion and contamination from radionuclides and also provide wear protection during plant operation and maintenance activities. The NRC staff's review covers protective coating systems used inside the containment for their suitability for and stability under design-basis accident (DBA) conditions, considering radiation and chemical effects. The NRC's acceptance criteria for protective coating systems are based on (1) 10 CFR Part 50, Appendix B, for the quality assurance requirements for the design, fabrication, and construction of safety-related SSCs and (2) Regulatory Guide 1.54, Revision 1, for application and performance monitoring of coatings in nuclear power plants. Specific review criteria are contained in SRP Section 6.1.2.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed EPU on protective coating systems and concludes that the licensee has appropriately addressed the impact of changes in conditions following a design-basis loss-of-coolant accident (DBLOCA) and their effects on the protective coatings. The NRC staff further concludes that the licensee has demonstrated that the protective coatings will continue to be acceptable following implementation of the proposed EPU and will continue to meet the requirements of 10 CFR Part 50, Appendix B. Therefore, the NRC staff finds the proposed EPU acceptable with respect to protective coatings systems.

2.1.6 Flow-Accelerated Corrosion

Regulatory Evaluation

Flow-accelerated corrosion (FAC) is a corrosion mechanism occurring in carbon steel components exposed to flowing single- or two-phase water. The components made from stainless steel are immune to FAC, and FAC is significantly reduced in components containing small amounts of chromium or molybdenum. The rates of material loss by FAC depend on velocity of flow, temperature, steam quality, oxygen content, and pH. During plant operation, control of these parameters is limited and the optimum conditions for minimizing FAC effects, in most cases, cannot be achieved. Loss of material by FAC will, therefore, occur. The NRC staff reviews the effects of the proposed EPU on FAC and the adequacy of the licensee's FAC program to predict the rate of loss so that repair or replacement of damaged components could be made before they reach critical thickness. The licensee's FAC program is based on NUREG-1344, GL 89-08, and the guidelines in Electric Power Research Institute (EPRI) Report NSAC-202L-R2. It consists of predicting loss of material using the CHECWORKS computer code, and visual inspection and volumetric examination of the affected components. The NRC's acceptance criteria are based on the structural evaluation of the minimum acceptable wall thickness for the components undergoing degradation by FAC.

Technical Evaluation

[Insert technical evaluation.]

Conclusions

The NRC staff has reviewed the licensee's evaluation of the effect of the proposed EPU on the FAC analysis for the plant and concludes that the licensee has adequately addressed changes in the plant operating conditions on the FAC analysis. Further, the NRC staff concludes that the licensee has demonstrated that the updated analyses will predict the loss of material by FAC and will ensure timely repair or replacement of degraded components following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to FAC.

2.1.7 Reactor Water Cleanup System

Regulatory Evaluation

The reactor water cleanup system (RWCS) provides a means for maintaining reactor water quality by filtration and ion exchange and provides a path for removal of reactor coolant when required. The NRC staff's review of the RWCS includes component design parameters for flow, temperature, pressure, heat removal capability, and impurity removal capability; and the instrumentation and process controls for proper system operation and isolation. The review consists of evaluating the adequacy of the plant's TSs in these areas. The NRC's acceptance criteria for the RWCS are based on (1) GDC-14 for ensuring the RCPB integrity, (2) GDC-60 for the capability of the RWCS to control the release of radioactive effluents to the environment, and (3) GDC-61 for appropriate confinement of fluids in the RWCS. Specific review criteria are contained in SRP Section 5.4.8.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed EPU on the RWCS and concludes that the licensee has adequately addressed changes in impurity levels and pressure and their effects on the RWCS. The NRC staff further concludes that the licensee has demonstrated that the RWCS will continue to be acceptable following implementation of the proposed EPU and will continue to meet the requirements of GDC-14, GDC-60, and GDC-61. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the RWCS.

[2.1.8 Additional Review Areas (Materials and Chemical Engineering)]

[Insert Regulatory Evaluation, Technical Evaluation, and Conclusion sections as necessary]

INSERT 2

FOR

SECTION 3.2 - BWR TEMPLATE SAFETY EVALUATION

2.2 Mechanical and Civil Engineering

2.2.1 Pipe Rupture Locations and Associated Dynamic Effects

Regulatory Evaluation

SSCs important to safety could be impacted by the pipe-whip dynamic effects of a pipe rupture. The NRC staff conducts a review of pipe rupture analyses to ensure that SSCs important to safety are adequately protected from the effects of pipe ruptures. The NRC staff's review covers (1) the implementation of criteria for defining pipe break and crack locations and configurations, (2) the implementation of criteria dealing with special features, such as augmented inservice inspection programs or the use of special protective devices such as pipe-whip restraints, (3) the pipe-whip dynamic analyses and results, including the jet thrust and impingement forcing functions and pipe-whip dynamic effects, and (4) the design adequacy of supports for SSCs provided to ensure that the intended design functions of the SSCs will not be impaired to an unacceptable level as a result of pipe-whip or jet impingement loadings. The NRC staff's review is focused on the effects that the proposed EPU may have on items (1) thru (4) above. The NRC's acceptance criteria are based on GDC-4 as related to SSCs important to safety being designed to accommodate the dynamic effects of a postulated pipe rupture. Specific review criteria are contained in SRP Section 3.6.2.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's evaluations related to determinations of rupture locations and associated dynamic effects and concludes that the licensee has adequately addressed the effects of the proposed EPU on them. The NRC staff further concludes that the licensee has demonstrated that SSCs important to safety will continue to meet the requirements of GDC-4 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the determination of rupture locations and dynamic effects associated with the postulated rupture of piping.

2.2.2 Pressure-Retaining Components and Component Supports

Regulatory Evaluation

The NRC staff's review concerns the structural integrity of pressure-retaining components (and their supports) designed in accordance with the American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (B&PV Code), Section III, Division 1, and GDCs 1, 2, 4, 14, and 15. The NRC staff's review is focused on the effects of the proposed EPU on the design input parameters and the design-basis loads and load combinations for normal operating, upset, emergency, and faulted conditions. The NRC staff's review covers (1) the analyses of flow-induced vibration and (2) the analytical methodologies, assumptions, ASME Code editions, and computer programs used for these analyses. The NRC staff's review also includes a comparison of the resulting stresses and cumulative fatigue usage factors (CUFs) against the code-allowable limits. The NRC's acceptance criteria are based on (1) 10 CFR 50.55a and GDC-1 as they relate to SSCs being designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed; (2) GDC-2 as it relates to SSCs important to safety being designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions; (3) GDC-4 as it relates to SSCs important to safety being designed to accommodate the effects of and to be compatible with the environmental conditions of normal and accident conditions; (4) GDC-14 as it relates to the RCPB being designed, fabricated, erected, and tested to have an extremely low probability of abnormal leakage, rapidly propagating failure, and gross rupture; and (5) GDC-15 as it relates to the RCS being designed with sufficient margin to ensure that the design conditions are not exceeded. Specific review criteria are contained in SRP Sections 3.9.1, 3.9.2, 3.9.3, and 5.2.1.1; and other guidance provided in Matrix 2 of RS-001.

Technical Evaluation

Nuclear Steam Supply System Piping, Components, and Supports

[Insert technical evaluation for Nuclear Steam Supply System piping, components, and supports.]

Balance-of-Plant Piping, Components, and Supports

[Insert technical evaluation for balance-of-plant piping, components, and supports.]

Reactor Vessel and Supports

[Insert technical evaluation for reactor vessel and supports.]

Control Rod Drive Mechanism

[Insert technical evaluation for control rod drive mechanism.]

Recirculation Pumps and Supports

[Insert technical evaluation for reactor coolant pumps and supports.]

Conclusion

The NRC staff has reviewed the licensee's evaluations related to the structural integrity of pressure-retaining components and their supports and concludes that the licensee has adequately addressed the effects of the proposed EPU on them. The NRC staff further concludes that the licensee has demonstrated that pressure-retaining components and their supports will continue to meet the requirements of 10 CFR 50.55a, GDC-1, GDC-2, GDC-4, GDC-14, and GDC-15 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the structural integrity of the pressure-retaining components and their supports.

2.2.3 Reactor Pressure Vessel Internals and Core Supports

Regulatory Evaluation

Reactor pressure vessel internals consist of all the structural and mechanical elements inside the reactor vessel, including core support structures. The NRC staff reviews the effects of the proposed EPU on the design input parameters and the design-basis loads and load combinations for the reactor internals for normal operation, upset, emergency, and faulted conditions. These include pressure differences and thermal effects for normal operation, transient pressure loads associated with LOCAs, and the identification of design transient occurrences. The NRC staff's review covers (1) the analyses of flow-induced vibration for safety-related and non-safety-related reactor internal components and (2) the analytical methodologies, assumptions, ASME Code editions, and computer programs used for these analyses. The NRC staff's review also includes a comparison of the resulting stresses and CUFs against the corresponding Code-allowable limits. The NRC's acceptance criteria are based on (1) GDC-1 and 10 CFR 50.55a for the design of reactor internals using quality standards commensurate with the importance of the safety functions to be performed; (2) GDC-2 for the design of reactor internals to withstand the effects of earthquakes without the loss of capability to perform their safety functions; (3) GDC-4 for the design of reactor internals to accommodate the effects of and to be compatible with the environmental conditions associated with normal operations, maintenance, testing, and postulated LOCAs; and (4) GDC-10 for the design of reactor internals with appropriate margin to assure that specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences. Specific review criteria are contained in SRP Sections 3.9.1, 3.9.2, 3.9.3, and 3.9.5; and other guidance provided in Matrix 2 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's evaluations related to the structural integrity of reactor internals and core supports and concludes that the licensee has adequately addressed the effects of the proposed EPU on them. The NRC staff further concludes that the licensee has demonstrated that the reactor internals and core supports will continue to meet the requirements of 10 CFR 50.55a, GDC-1, GDC-2, GDC-4, and GDC-10 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the design of the reactor internal and core supports.

2.2.4 Safety-Related Valves and Pumps

Regulatory Evaluation

The NRC's staff's review includes certain safety-related pumps and valves typically designated as Class 1, 2, or 3 under Section III of the ASME B&PV Code and within the scope of Section XI of the ASME B&PV Code and the ASME Operations and Maintenance (O&M) Code, as applicable. The NRC staff's review focuses on the effects of the proposed EPU on the required functional performance of the valves and pumps. The review also covers any impacts that the proposed EPU may have on the licensee's motor-operated valve (MOV) programs related to GL 89-10, GL 96-05, and GL 95-07. The NRC staff also evaluates the licensee's consideration of lessons learned from the MOV program and the application of those lessons learned to other safety-related power-operated valves. The NRC's acceptance criteria are based on (1) GDC-1 for testing components important to safety to quality standards commensurate with the importance of the safety functions to be performed; (2) GDC-37, GDC-40, GDC-43, and GDC-46 for periodic functional testing of the emergency core cooling system, the containment heat removal system, the containment atmospheric cleanup systems, and the cooling water system, respectively, to ensure the leak-tight integrity and performance of their active components; (3) GDC-54 for piping systems penetrating containment being designed with the capability to periodically test the operability of the isolation and determine valve leakage acceptability; and (4) 10 CFR 50.55a(f) for including pumps and valves whose function is required for safety in the inservice testing program to verify operational readiness by periodic testing. Specific review criteria are contained in SRP Sections 3.9.3 and 3.9.6; and other guidance provided in Matrix 2 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's evaluations related to the functional performance of safety-related valves and pumps and concludes that the licensee has adequately addressed the effects of the proposed EPU on them. The NRC staff further concludes that the licensee has adequately evaluated the effects of the proposed EPU on its MOV programs related to GL 89-10, GL 96-05, and GL 95-07, and the lessons learned from those programs to other safety-related power-operated valves. Based on this, the NRC staff concludes that the licensee has demonstrated that safety-related valves and pumps will continue to meet the requirements of GDC-1, GDC-37, GDC-40, GDC-43, GDC-46, GDC-54, and 10 CFR 50.55a(f) following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to safety-related valves and pumps.

2.2.5 Seismic and Dynamic Qualification of Mechanical and Electrical Equipment

Regulatory Evaluation

Mechanical and electrical equipment covered by this section includes equipment associated with systems that are essential to emergency reactor shutdown, containment isolation, reactor core cooling, and containment and reactor heat removal. Equipment associated with systems essential in preventing significant release of radioactive materials to the environment are also covered by this section. The NRC staff's review focuses on the effects of the proposed EPU on the qualification of the equipment to withstand seismic events and the dynamic effects associated pipe-whip and jet impingement forces. The primary input motions due to the safe shutdown earthquake (SSE) are not affected by an EPU. The NRC's acceptance criteria are based on (1) GDC-1 and GDC-30 for qualifying equipment to appropriate quality standards commensurate with the importance of the safety functions to be performed; (2) GDC-2 and Appendix A to 10 CFR Part 100 for qualifying equipment to withstand the effects of natural phenomena, such as earthquakes; (3) GDC-4 for qualifying equipment to withstand the dynamic effects associated with external missiles and internally generated missiles, pipe whip, and jet impingement forces; (4) GDC-14 for qualifying equipment associated with the RCPB to ensure an extremely low probability of abnormal leakage, rapidly propagating failure, and gross rupture; and (5) Appendix B to 10 CFR Part 50 for the quality assurance requirements for qualification of equipment. Specific review criteria are contained in SRP Section 3.10.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's evaluations of the effects of the proposed EPU on the qualification of mechanical and electrical equipment and concludes that the licensee has (1) adequately addressed the effects of the proposed EPU on this equipment and (2) demonstrated that the equipment will continue to meet the requirements of GDCs 1, 2, 4, 14, and 30; 10 CFR Part 100, Appendix A; and 10 CFR Part 50, Appendix B, following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the qualification of the mechanical and electrical equipment.

[2.2.6 Additional Review Areas (Mechanical and Civil Engineering)]

[Insert Regulatory Evaluation, Technical Evaluation, and Conclusion sections as necessary]

INSERT 3

FOR

SECTION 3.2 - BWR TEMPLATE SAFETY EVALUATION

2.3 Electrical Engineering

2.3.1 Environmental Qualification of Electrical Equipment

Regulatory Evaluation

Environmental qualification (EQ) of electrical equipment involves demonstrating that the equipment is capable of performing its safety function under significant environmental stresses which could result from DBAs. The NRC staff's review is focused on the effects of the proposed EPU on the environmental conditions that the electrical equipment will be exposed to during normal operation, anticipated operational occurrences, and accidents. The NRC staff's review is conducted to ensure that the electrical equipment will continue to be capable of performing its safety functions following implementation of the proposed EPU. The NRC's acceptance criteria for EQ of electrical equipment are based on 10 CFR 50.49 as it relates to the qualification of electrical equipment important to safety that is located in a harsh environment. Specific review criteria are contained in SRP Section 3.11.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the EQ of electrical equipment and concludes that the licensee has adequately addressed the effects of the proposed EPU on the environmental conditions for and the qualification of the electrical equipment. The NRC staff further concludes that the electrical equipment will continue to meets the relevant requirements of 10 CFR 50.49 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the EQ of electrical equipment.

2.3.2 Offsite Power System

Regulatory Evaluation

The offsite power system includes two or more physically independent circuits capable of operating independently of the onsite standby power sources. The NRC staff's review covers the descriptive information, analyses, and referenced documents for the offsite power system; and the stability studies for the electrical transmission grid. The NRC staff's review is focused on the requirement that loss of the nuclear unit, the largest operating unit on the grid, or the most critical transmission line will not result in the loss of offsite power to the plant following implementation of the proposed EPU. The NRC's acceptance criteria for offsite power systems are based on GDC-17. Specific review criteria are contained in SRP Sections 8.1 and 8.2, Appendix A to SRP Section 8.2, and Branch Technical Positions (BTPs) PSB-1 and ICSB-11.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the offsite power system and concludes that the offsite power system will continue to meet the requirements of GDC-17 following implementation of the proposed EPU. Adequate physical and electrical separation exists and the offsite power system has the capacity and capability to supply power to all safety loads and other required equipment. The NRC staff further concludes that the impact of the proposed EPU on grid stability is insignificant. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the offsite power system.

2.3.3 AC Onsite Power System

Regulatory Evaluation

The ac onsite power system includes those standby power sources, distribution systems, and auxiliary supporting systems provided to supply power to safety-related equipment. The NRC staff's review covers the descriptive information, analyses, and referenced documents for the ac onsite power system. The NRC's acceptance criteria for the ac onsite power system are based on GDC-17 for the capability of the ac onsite power system to perform its intended functions during all plant operating and accident conditions. Specific review criteria are contained in SRP Sections 8.1 and 8.3.1.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the ac onsite power system and concludes that the licensee has adequately accounted for the effects of the proposed EPU on the system's functional design. The NRC staff further concludes that the ac onsite power system will continue to meet the requirements of GDC-17 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the ac onsite power system.

2.3.4 DC Onsite Power System

Regulatory Evaluation

The dc onsite power system includes the dc power sources and their distribution and auxiliary supporting systems that are provided to supply motive or control power to safety-related equipment. The NRC staff's review covers the information, analyses, and referenced documents for the dc onsite power system. The NRC's acceptance criteria for the dc onsite power system are based on GDC-17 for the capability of the dc onsite power system to facilitate the functioning of SSCs important to safety. Specific review criteria are contained in SRP Sections 8.1 and 8.3.2

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the dc onsite power system and concludes that the licensee has adequately accounted for the effects of the proposed EPU on the system's functional design. The NRC staff further concludes that the dc onsite power system will continue to meet the requirements of GDC-17 following implementation of the proposed EPU. Adequate physical and electrical separation exists and the system has the capacity and capability to supply power to all safety loads and other required equipment. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the dc onsite power system.

2.3.5 Station Blackout

Regulatory Evaluation

Station blackout (SBO) refers to a complete loss of ac electric power to the essential and nonessential switchgear buses in a nuclear power plant. SBO involves the loss of offsite power concurrent with a turbine trip and failure of the onsite emergency ac power system. SBO does not include the loss of available ac power to buses fed by station batteries through inverters or the loss of power from "alternate ac sources" (AACs). The NRC staff's review focuses on the impact of the proposed EPU on the plant's ability to cope with and recover from a SBO event for the period of time established in the plant's licensing basis. The NRC's acceptance criteria for SBO are based on 10 CFR 50.63. Specific review criteria are contained in SRP Sections 8.1 and Appendix B to SRP Section 8.2; and other guidance provided in Matrix 3 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the plant's ability to cope with and recover from a SBO event for the period of time established in the plant's licensing basis. The NRC staff concludes that the licensee has adequately evaluated the effects of the proposed EPU on SBO and demonstrated that the plant will continue to meet the requirements of 10 CFR 50.63 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to SBO.

[2.3.6 Additional Review Areas (Electrical Engineering)]

[Insert Regulatory Evaluation, Technical Evaluation, and Conclusion sections as necessary]

INSERT 4

FOR

SECTION 3.2 - BWR TEMPLATE SAFETY EVALUATION

2.4 Instrumentation and Controls

2.4.1 Reactor Protection, Safety Features Actuation, and Control Systems

Regulatory Evaluation

Instrumentation and control systems are provided (1) to control plant processes having a significant impact on plant safety, (2) to initiate the reactivity control system (control rods), (3) to initiate the engineered safety features (ESF) systems and essential auxiliary supporting systems, and (4) for use to achieve and maintain a safe shutdown condition of the plant. Diverse instrumentation and control systems and equipment are provided for the express purpose of protecting against potential common-mode failures of instrumentation and control protection systems. The NRC staff conducts a review of the reactor trip system, engineered safety feature actuation system (ESFAS), safe shutdown systems, control systems, and diverse instrumentation and control systems for the proposed EPU to ensure that the systems and any changes required for the proposed EPU are adequately designed such that the systems continue to meet their safety functions. The NRC staff's review is also conducted to ensure that failures of the systems do not affect safety functions. The NRC's acceptance criteria related to the quality of design of protection and control systems are based on 10 CFR 50.55a(a)(1), 10 CFR 50.55a(h), and GDCs 1, 4, 13, 19, 20, 21, 22, 23, and 24. Specific review criteria are contained in SRP Sections 7.0, 7.2, 7.3, 7.4, 7.7, and 7.8.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's application related to the effects of the proposed EPU on the functional design of the reactor trip system, ESFAS, safe shutdown system, and control systems. The NRC staff concludes that the licensee has adequately addressed the effects of the proposed EPU on these systems and that the changes that are required to achieve the proposed EPU are consistent with the plant's licensing basis. The NRC staff further concludes that the systems will continue to meet the requirements of 10 CFR 50.55a(a)(1), 10 CFR 50.55(a)(h), and GDCs 1, 4, 13, 19, 20, 21, 22, 23, and 24. Therefore, the NRC staff finds the licensee's proposed EPU acceptable with respect to instrumentation and controls.

[2.4.2 Additional Review Areas (Instrumentation and Controls)]

[Insert Regulatory Evaluation, Technical Evaluation, and Conclusion sections as necessary]

INSERT 5

FOR

SECTION 3.2 - BWR TEMPLATE SAFETY EVALUATION

2.5 Plant Systems

2.5.1 Internal Hazards

2.5.1.1 Flooding

2.5.1.1.1 Flood Protection

Regulatory Evaluation

The NRC staff conducts its review in the area of flood protection to ensure that SSCs important to safety are protected from flooding. The NRC staff's review covers flooding of SSCs important to safety from internal sources, such as those caused by failures of tanks and vessels. The NRC staff's review focuses on increases of fluid volumes in tanks and vessels assumed in flooding analyses to assess the impact of any additional fluid on the flooding protection that is provided. The NRC's acceptance criteria for flood protection are based on GDC-2. Specific review criteria are contained in SRP Section 3.4.1.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the proposed changes in fluid volumes in tanks and vessels for the proposed EPU. The NRC staff concludes that SSCs important to safety will continue to be protected from flooding and will continue to meet the requirements of GDC-2 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to flood protection.

2.5.1.1.2 Equipment and Floor Drains

Regulatory Evaluation

The function of the equipment and floor drainage system (EFDS) is to assure that waste liquids, valve and pump leakoffs, and tank drains are directed to the proper area for processing or disposal. The EFDS is designed to handle the volume of leakage expected, prevent a backflow of water that might result from maximum flood levels to areas of the plant containing safety-related equipment, and protect against the potential for inadvertent transfer of contaminated fluids to a noncontaminated drainage system. The NRC staff's review of the EFDS includes the collection and disposal of liquid effluents outside containment.
The NRC staff's review is focused on any changes in fluid volumes or pump capacities that are required for the proposed EPU and are not consistent with previous assumptions with respect to floor drainage considerations. The NRC's acceptance criteria for the EFDS are based on GDCs 2 and 4 for the capability of the EFDS to withstand the effects of earthquakes and to be compatible with the environmental conditions (flooding) associated with normal operation, maintenance, testing, and postulated accidents (pipe failures and tank ruptures). Specific review criteria are contained in SRP Section 9.3.3.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the EFDS and concludes that the licensee has adequately accounted for the plant changes resulting in increased water volumes and larger capacity pumps or piping systems. The NRC staff concludes that the EFDS has sufficient capacity to (1) handle the additional expected leakage resulting from the plant changes, (2) prevent the backflow of water to areas with safety-related equipment, and (3) ensure that contaminated fluids are not transferred to noncontaminated drainage systems. Based on this, the NRC staff concludes that the EFDS will continue to meet the requirements of GDCs 2 and 4 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the EFDS.

2.5.1.1.3 Circulating Water System

Regulatory Evaluation

The circulating water system (CWS) provides a continuous supply of cooling water to the main condenser to remove the heat rejected by the turbine cycle and auxiliary systems. The NRC staff's review of the CWS focuses on changes in flooding analyses that are necessary due to increases in fluid volumes or installation of larger capacity pumps or piping required to accommodate the proposed EPU. The NRC's acceptance criteria for the CWS are based on GDC-4 for the effects of flooding of safety-related areas due to leakage from the CWS and the effects of malfunction or failure of a component or piping of the CWS on the functional performance capabilities of safety-related SSCs. Specific review criteria are contained in SRP Section 10.4.5.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the modifications to the CWS and concludes that the licensee has adequately evaluated these modifications. The NRC staff concludes that, consistent with the requirements of GDC-4, the increased volumes of fluid leakage that could potentially result from these modifications would not result in the failure of safety-related SSCs following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the CWS.

2.5.1.2 Missile Protection

2.5.1.2.1. Internally Generated Missiles

Regulatory Evaluation

The NRC staff's review concerns missiles that could result from in-plant component overspeed failures and high-pressure system ruptures. The NRC staff's review of potential missile sources covers pressurized components and systems, and high-speed rotating machinery. The NRC staff's review is conducted to ensure that safety-related SSCs are adequately protected from internally generated missiles. In addition, if safety-related SSCs are located in areas containing non-safety-related SSCs, the NRC staff reviews the non-safety-related SSCs to ensure that their failure will not preclude the intended safety function of the safety-related SSCs. The NRC staff's review focuses on any increases in system pressures or component overspeed conditions that could result during plant operation, anticipated operational occurrences, or changes in existing system configurations such that missile barrier considerations could be affected. The NRC's acceptance criteria for the protection SSCs important to safety against the effects of internally generated missiles that may result from equipment failures are based on GDC-4. Specific review criteria are contained in SRP Sections 3.5.1.1 and 3.5.1.2.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the changes in system pressures and configurations that are required for the proposed EPU and concludes that SSCs important to safety will continue to be protected from internally generated missiles and will continue to meet the requirements of GDC-4 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to internally generated missiles.

2.5.1.2.2 Turbine Generator

Regulatory Evaluation

The turbine control system, steam inlet stop and control valves, low pressure turbine steam intercept and inlet control valves, and extraction steam control valves control the speed of the turbine under normal and abnormal conditions, and are thus related to the overall safe operation of the plant. The NRC staff's review of the turbine generator focuses on the effects of the proposed EPU on the turbine overspeed protection features to ensure that a turbine overspeed condition above the design overspeed is very unlikely. The NRC's acceptance criteria for the turbine generator are based on GDC-4 for protection of SSCs important to safety from the effects of turbine missiles by providing a turbine overspeed protection system (with suitable redundancy) to minimize the probability of generating turbine missiles. Specific review criteria are contained in SRP Section 10.2.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the turbine generator and concludes that the licensee has adequately accounted for the effects of changes in plant conditions on turbine overspeed. The NRC staff concludes that the turbine generator will continue to provide adequate turbine overspeed protection to minimize the probability of generating turbine missiles and will continue to meet the requirements of GDC-4 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the turbine generator.

2.5.1.3 Pipe Failures

Regulatory Evaluation

The NRC staff conducts a review of the plant design for protection from piping failures outside containment to ensure that (1) such failures would not cause the loss of needed functions of safety-related systems and (2) the plant could be safely shut down in the event of such failures. The NRC staff's review of pipe failures includes high and moderate energy fluid system piping located outside of containment. The NRC staff's review focuses on the effects of pipe failures on the resulting environmental conditions, control room habitability, and access to areas important to safe control of postaccident operations where the consequences are not bounded by previous analyses. The NRC's acceptance criteria for pipe failures are based on GDC-4 for SSCs important to safety being designed to accommodate the dynamic effects of postulated pipe ruptures, including the effects of pipe whipping and discharging fluids. Specific review criteria are contained in SRP Section 3.6.1.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the changes that are required for the proposed EPU and the licensee's proposed operation of the plant, and concludes that SSCs important to safety will continue to be protected from postulated piping failures in fluid systems outside containment and will continue to meet the requirements of GDC-4 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to protection against postulated piping failures in fluid systems outside containment.

2.5.1.4 Fire Protection

Regulatory Evaluation

The purpose of the fire protection program (FPP) is to provide assurance, through a defense-in-depth design, that a fire will not prevent the performance of necessary safe plant shutdown functions and will not significantly increase the risk of radioactive releases to the environment. The NRC staff's review focuses on the effects of the increased decay heat on the plant's safe shutdown analysis to ensure that SSCs required for the safe shutdown of the plant are protected from the effects of the fire and continue to be able to achieve and maintain safe shutdown following a fire. The NRC's acceptance criteria for the FPP are based on (1) 10 CFR 50.48 and associated Appendix R to 10 CFR Part 50 for the development of a fire protection plan to ensure the capability to safely shut down the plant; (2) GDC-3 for fire prevention, the design and operation of fire detection and suppression systems, and administrative controls provided to protect SSCs important to safety; and (3) GDC-5 for fire protection for shared safety-related SSCs to assure the ability to perform their intended safety function. Specific review criteria are contained in SRP Section 9.5.1, as supplemented by the guidance provided in Attachment 3 to Matrix 5 of Section 2.1 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's fire-related safe shutdown assessment and concludes that the licensee has adequately accounted for the effects of the increased decay heat on the ability of the required systems to achieve and maintain safe shutdown conditions. The NRC staff further concludes that the FPP will continue to meet the requirements of 10 CFR 50.48, 10 CFR Part 50, Appendix R, and GDCs 3 and 5 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to fire protection.

2.5.2 Fission Product Control

2.5.2.1 Fission Product Control Systems and Structures

Regulatory Evaluation

The NRC staff's review for fission product control systems and structures covers the basis for developing the mathematical model for DBLOCA dose computations, the values of key parameters, the applicability of important modeling assumptions, and the functional capability of ventilation systems used to control fission product releases. The NRC staff's review primarily focuses on any adverse effects that the proposed EPU may have on the assumptions used in the analyses for control of fission products. The NRC's acceptance criteria are based on GDC-41 for the containment atmosphere cleanup system being designed to control fission product releases to the environment following postulated accidents. Specific review criteria are contained in SRP Section 6.5.3.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on fission product control systems and structures. The NRC staff concludes that the licensee has adequately accounted for the increase of fission products and changes in expected environmental conditions that would result from the proposed EPU, and the NRC staff further concludes that the fission product control systems and structures will continue to provide adequate fission product removal in postaccident environments following implementation of the proposed EPU. Based on this, the NRC staff also concludes that the fission product control systems and structures will continue to meet the requirements of GDC-41. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the fission product control systems and structures.

2.5.2.2 Main Condenser Evacuation System

Regulatory Evaluation

The main condenser evacuation system (MCES) generally consists of two subsystems: (1) the "hogging" or startup system which initially establishes main condenser vacuum and (2) the system which maintains condenser vacuum once it has been established. The NRC staff's review focuses on modifications to the system that may affect gaseous radioactive material handling and release assumptions, and design features to preclude the possibility of an explosion (if the potential for explosive mixtures exists). The NRC's acceptance criteria for the MCES are based on (1) GDC-60 for the MCES design for the control of releases of radioactive materials to the environment and (2) GDC-64 for the MCES design for the monitoring of releases of radioactive materials to the environment. Specific review criteria are contained in SRP Section 10.4.2.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment of required changes to the MCES and concludes that the licensee has adequately evaluated these changes. The NRC staff concludes that the MCES will continue to maintain its ability to control and provide monitoring for releases of radioactive materials to the environment following implementation of the proposed EPU. The NRC also concludes that the MCES will continue meet the requirements of GDCs 60 and 64. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the MCES.

2.5.2.3 Turbine Gland Sealing System

Regulatory Evaluation

The turbine gland sealing system is provided to control the release of radioactive material from steam in the turbine to the environment. The NRC staff reviews changes to the turbine gland sealing system with respect to factors that may affect gaseous radioactive material handling (e.g., source of sealing steam, system interfaces, and potential leakage paths). The NRC's acceptance criteria for the turbine gland sealing system are based on (1) GDC-60 for the turbine gland sealing system design for the control of releases of radioactive materials to the environment and (2) GDC-64 for the turbine gland sealing system design for the monitoring of releases of radioactive materials to the environment. Specific review criteria are contained in SRP Section 10.4.3.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment of required changes to the turbine gland sealing system and concludes that the licensee has adequately evaluated these changes. The NRC staff concludes that the turbine gland sealing system will continue to maintain its ability to control and provide monitoring for releases of radioactive materials to the environment consistent with GDCs 60 and 64. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the turbine gland sealing system.

2.5.2.4 Main Steam Isolation Valve Leakage Control System

Regulatory Evaluation

Redundant quick-acting isolation valves are provided on each main steamline. The leakage control system is designed to reduce the amount of direct, untreated leakage from the main steam isolation valves (MSIVs) when isolation of the primary system and containment is required. The NRC staff's review of the MSIV leakage control system focuses on the effects of the proposed EPU on the amount of leakage assumed to occur. The NRC's acceptance criteria for the MSIV leakage control system are based on GDC-54 for the capability for leak detection and isolation. Specific review criteria are contained in SRP Section 6.7.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment related to the MSIV leakage control system and finds that the licensee has adequately accounted for the effects of the proposed EPU on the assumed leakage through the MSIVs. The NRC staff further concludes that the leakage control system will continue to reliably detect and isolate the leakage, as required by GDC-54. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the MSIV leakage control system.

2.5.3 Component Cooling and Decay Heat Removal

2.5.3.1 Spent Fuel Pool Cooling and Cleanup System

Regulatory Evaluation

The spent fuel pool provides wet storage of spent fuel assemblies. The safety function of the spent fuel pool cooling and cleanup system is to cool the spent fuel assemblies and keep the spent fuel assemblies covered with water during all storage conditions. The NRC staff's review for the proposed EPU focuses on the effects of the proposed EPU on the capability of the system to provide adequate cooling to the spent fuel during all operating and accident conditions. The NRC's acceptance criteria for the spent fuel pool cooling and cleanup system are based on (1) GDC-5 for shared systems and components important to safety being capable of performing required safety functions, (2) GDC-44 for the capability to transfer heat loads from safety-related SSCs to a heat sink under both normal operating and accident conditions, and (3) GDC-61 for the RHR capability and measures to prevent a significant loss of fuel storage coolant inventory under accident conditions. Specific review criteria are contained in SRP Section 9.1.3, as supplemented by the guidance provided in Attachment 2 to Matrix 5 of Section 2.1 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment related to the spent fuel pool cooling and cleanup system and concludes that the licensee has adequately accounted for the effects of the proposed EPU on the spent fuel pool cooling function of the system. Based on this review, the NRC staff concludes that the spent fuel pool cooling and cleanup system will continue to provide sufficient cooling capability to cool the spent fuel pool following implementation of the proposed EPU and will continue to meet the requirements of GDCs 5, 44, and 61. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the spent fuel pool cooling and cleanup system.

2.5.3.2 Station Service Water System

Regulatory Evaluation

The station service water system (SWS) provides essential cooling to safety-related equipment and may also provide cooling to non-safety-related auxiliary components that are used for normal plant operation. The NRC staff's review covers the characteristics of the station SWS components with respect to their functional performance as affected by adverse operational (i.e., water hammer) requirements, abnormal operational requirements, and accident conditions (e.g., a LOCA with the loss of offsite power (LOOP)). The NRC staff's review focuses on the additional heat load that results from the proposed EPU. The NRC's acceptance criteria are based on (1) GDC-4 for dynamic effects associated with flow instabilities and loads (e.g., water hammer) during normal plant operation, as well as during upset or accident conditions; (2) GDC-5 for the capability of shared systems and components important to safety to perform their required safety functions; and (3) GDC-44 for transferring heat from SSCs important to safety to an ultimate heat sink. Specific review criteria are contained in SRP Section 9.2.1, as supplemented by GL 89-13 and GL 96-06.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment related to the effects of the proposed EPU on the station SWS and concludes that the licensee has adequately accounted for the increased heat loads from the proposed EPU on system performance. The NRC staff concludes that the station SWS will continue to be protected from the dynamic effects associated with flow instabilities and provide sufficient cooling for SSCs important to safety following implementation of the proposed EPU. Therefore, the NRC staff has determined that the station SWS will continue to meet the requirements of GDCs 4, 5, and 44. Based on the above, the NRC staff finds the proposed EPU acceptable with respect to the station SWS.

2.5.3.3 Reactor Auxiliary Cooling Water Systems

Regulatory Evaluation

The NRC staff's review covers reactor auxiliary cooling water systems that are required for (1) safe shutdown during normal operations, anticipated operational occurrences, and for mitigating the consequences of accident conditions, or (2) preventing the occurrence of an accident. These systems include closed-loop auxiliary cooling water systems for reactor system components, reactor shutdown equipment, ventilation equipment, and components of the ECCS. The NRC staff's review covers the capability of the auxiliary cooling water systems to provide adequate cooling water to safety-related ECCS components and reactor auxiliary equipment for all planned operating conditions. Emphasis is placed on the cooling water systems for safety-related components such as ECCS equipment, ventilation equipment, and reactor shutdown equipment. The NRC staff's review focuses on the additional heat load that results from the proposed EPU. The NRC's acceptance criteria for the reactor auxiliary CWS are based on (1) GDC-4 for dynamic effects associated with flow instabilities and attendant loads (i.e., water hammer) during normal plant operation, as well as during upset or accident conditions; (2) GDC-5 for shared systems and components important to safety being capable of performing required safety functions; and (3) GDC-44 for the capability to transfer heat loads from safety-related SSCs to a heat sink under both normal operating and accident conditions. Specific review criteria are contained in SRP Section 9.2.2, as supplemented by GL 89-13 and GL 96-06.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the reactor auxiliary cooling water systems and concludes that the licensee has adequately accounted for the increased heat loads from the proposed EPU on system performance. The NRC staff concludes that the reactor auxiliary cooling water systems will continue to be protected from the dynamic effects associated with flow instabilities and provide sufficient cooling for SSCs important to safety following implementation of the proposed EPU. Therefore, the NRC staff has determined that the reactor auxiliary cooling water systems will continue to meet the requirements of GDCs 4, 5, and 44. Based on the above, the NRC staff finds the proposed EPU acceptable with respect to the reactor auxiliary cooling water systems.

2.5.3.4 Ultimate Heat Sink

Regulatory Evaluation

The ultimate heat sink (UHS) is the source of cooling water provided to dissipate reactor decay heat and essential cooling system heat loads after a normal reactor shutdown or a shutdown following an accident. The NRC staff's review is focused on the impact that the proposed EPU has on the decay heat removal capability of the UHS. Additionally, the NRC staff's review includes evaluation of the design-basis UHS temperature limit determination to confirm that post-licensing data trends (e.g., air and water temperatures, humidity, wind speed, water volume) do not establish more severe conditions than previously assumed. The NRC's acceptance criteria for the UHS are based on (1) GDC-5 for shared systems and components important to safety being capable of performing required safety functions, and (2) GDC-44 for the capability to transfer heat loads from safety-related SSCs to the heat sink under both normal operating and accident conditions. Specific review criteria are contained in SRP Section 9.2.5.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the information that was provided by the licensee for addressing the effects that the proposed EPU would have on the UHS safety function, including the licensee's validation of the design-basis UHS temperature limit based on post-licensing data. Based on the information that was provided, the NRC staff concludes that the proposed EPU will not compromise the design-basis safety function of the UHS, and that the UHS will continue to satisfy the requirements of GDCs 5 and 44. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the UHS.

2.5.4 Balance-of-Plant Systems

2.5.4.1. Main Steam

Regulatory Evaluation

The main steam supply system (MSSS) transports steam from the nuclear steam supply system to the power conversion system and various safety-related and non-safety-related auxiliaries. The NRC staff's review focuses on the effects of the proposed EPU on the system's capability to transport steam to the power conversion system, provide heat sink capacity, supply steam to drive safety system pumps, and withstand adverse dynamic loads (e.g., water steam hammer resulting from rapid valve closure and relief valve fluid discharge loads). The NRC's acceptance criteria for the MSSS are based on (1) GDC-4 for safety-related portions of the system being capable of withstanding the effects of external missiles and internally generated missiles, pipe whip, and jet impingement forces associated with pipe breaks; and (2) GDC-5 for the capability of shared systems and components important to safety to perform required safety functions. Specific review criteria are contained in SRP Section 10.3.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the MSSS and concludes that the licensee has adequately accounted for the effects of changes in plant conditions on the design of the MSSS. The NRC staff concludes that the MSSS will maintain its ability to transport steam to the power conversion system, provide heat sink capacity, supply steam to steam-driven safety pumps, and withstand steam hammer. The NRC staff further concludes that the MSSS will continue to meet the requirements of GDCs 4 and 34. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the MSSS.

2.5.4.2 Main Condenser

Regulatory Evaluation

The main condenser (MC) system is designed to condense and deaerate the exhaust steam from the main turbine and provide a heat sink for the turbine bypass system. For BWRs without an MSIV leakage control system, the MC system may also serve an accident mitigation function to act as a holdup volume for the plateout of fission products leaking through the MSIVs following core damage. The NRC staff's review focuses on the effects of the proposed EPU on the steam bypass capability with respect to load rejection assumptions, and on the ability of the MC system to withstand the blowdown effects of steam from the turbine bypass system. The NRC's acceptance criteria for the MC system are based on GDC-60 such that failures in the design of the system are not allowed to result in excessive releases of radioactivity to the environment or in unacceptable condensate quality, or in flooding of areas housing safety-related equipment. Specific review criteria are contained in SRP Section 10.4.1.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the MC system and concludes that the licensee has adequately accounted for the effects of changes in plant conditions on the design of the MC system. The NRC staff concludes that the MC system will continue to maintain its ability to withstand the blowdown effects of the steam from the turbine bypass system and thereby continue to meet GDC-60 for prevention of the consequences of failures in the system. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the MC system.

2.5.4.3 Turbine Bypass

Regulatory Evaluation

The turbine bypass system (TBS) is designed to discharge a stated percentage of rated main steam flow directly to the MC system, bypassing the turbine. This steam bypass enables the plant to take step-load reductions up to the TBS capacity without the reactor or turbine tripping. The system is also used during startup and shutdown to control reactor pressure. For a BWR without an MSIVLCS, the TBS could also provide an accident mitigation function. A TBS, along with the MSSS and MC system, may be credited for mitigating the effects of MSIV leakage during a LOCA by the holdup and plateout of fission products. The NRC staff's review for the TBS focuses on the effects that the proposed EPU have on load rejection capability, analysis of postulated system piping failures, and the consequences of inadvertent TBS operation. The NRC's acceptance criteria for the TBS are based on (1) GDC-4 for pipe break or malfunction of the TBS not adversely affecting essential SSCs, and (2) GDC-34 for the ability to use the system for shutting down the plant during normal operations. Specific review criteria are contained in SRP Section 10.4.4.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the TBS. The NRC staff concludes that the licensee has adequately accounted for the effects of changes in plant conditions on the design of the TBS. The NRC staff concludes that the TBS will continue to mitigate the effects of MSIV leakage during a LOCA and provide a means for shutting down the plant during normal operations. The NRC staff further concludes that TBS failures will not adversely affect essential SSCs. Based on this, the NRC staff concludes that the TBS will continue to meet GDCs 4 and 34. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the TBS.

2.5.4.4 Condensate and Feedwater

Regulatory Evaluation

The condensate and feedwater system (CFS) provides feedwater at the required temperature, pressure, and flow rate to the reactor. The only part of the CFS classified as safety-related is the feedwater piping from the nuclear steam supply system up to and including the outermost containment isolation valve. The NRC staff's review focuses on how the proposed EPU affects previous analyses and considerations with respect to the capability of the CFS to supply adequate feedwater during plant operation and shutdown, and isolate components, subsystems, and piping in order to preserve the system's safety function. The NRC's acceptance criteria for the CFS are based on (1) GDC-4 for the dynamic effects associated with possible fluid flow instabilities (e.g., water hammers) during normal plant operation, as well as during upset or accident conditions; (2) GDC-5 for the capability of shared systems and components important to safety to perform required safety functions; and (3) GDC-44 for satisfying feedwater flow requirements and system isolation considerations. Specific review criteria are contained in SRP Section 10.4.7.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the CFS and concludes that the licensee has adequately accounted for the effects of changes in plant conditions on the design of the CFS. The NRC staff concludes that the CFS will continue to maintain its ability to satisfy feedwater requirements for normal operation and shutdown, withstand water hammer, maintain isolation capability in order to preserve the system safety function, and not cause failure of safety-related SSCs. The NRC staff further concludes that the CFS will continue to meet the requirements of GDCs 4, 5, and 44. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the CFS.

2.5.5 Waste Management Systems

2.5.5.1. Gaseous Waste Management Systems

Regulatory Evaluation

The gaseous waste management systems involve the gaseous radwaste system, which deals with the management of radioactive gases collected in the offgas system or the waste gas storage and decay tanks. In addition, it involves the management of condenser air removal system; gland seal exhaust and mechanical vacuum pump operation exhaust; and building ventilation system exhausts. The NRC staff's review is focused on the effects that the proposed EPU have on previous analyses and considerations related to the gaseous waste management systems' design criteria, methods of treatment, expected releases, principal parameters used in calculating the releases of radioactive materials in gaseous effluents, and design features to preclude the possibility of an explosion if the potential for explosive mixtures exists. The NRC's acceptance criteria for gaseous waste management systems are based on (1) 10 CFR 20.1302 for radioactivity in effluents; (2) GDC-3 for providing protection for gaseous waste handling and treatment systems from the effects of an explosive mixture of hydrogen and oxygen; (3) GDC-60 for designing the gaseous waste management systems to control releases of radioactive materials to the environment; (4) GDC-61 for radioactivity control in gaseous waste management systems associated with fuel storage and handling areas; and (5) 10 CFR Part 50 Appendix I, Sections II.B., II.C., and II.D., for the numerical guides for design objectives and limiting conditions for operation to meet the "as low as is reasonably achievable" (ALARA) criterion. Specific review criteria are contained in SRP Section 11.3.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment related to the gaseous waste management systems. The NRC staff concludes that the licensee has adequately accounted for the effects of the increase in fission product and amount of gaseous waste on the abilities of the systems to control releases of radioactive materials and preclude the possibility of an explosion if the potential for explosive mixtures exists. The NRC staff finds that the gaseous waste management systems will continue to meet their design functions following implementation of the proposed EPU. The NRC staff further concludes that the licensee has demonstrated that the gaseous waste management systems will continue to meet the requirements of 10 CFR 20.1302, GDCs 3, 60, and 61, and 10 CFR Part 50, Appendix I, Sections II.B, II.C, and II.D. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the gaseous waste management systems.

2.5.5.2 Liquid Waste Management Systems

Regulatory Evaluation

The NRC staff's review for liquid waste management systems is focused on the effects that the proposed EPU have on previous analyses and considerations related to the liquid waste management systems' design, design objectives, design criteria, methods of treatment, expected releases, and principal parameters used in calculating the releases of radioactive materials in liquid effluents. The NRC's acceptance criteria for the liquid waste management systems are based on (1) 10 CFR 20.1302 for radioactivity in effluents to unrestricted areas; (2) GDC-60 for the liquid waste management systems being designed to control releases of radioactive materials to the environment; (3) GDC-61 for the liquid waste management systems being designed to ensure adequate safety under normal and postulated accident conditions; and (4) 10 CFR Part 50, Appendix I, Sections II.A and II.D for the numerical guides for dose design objectives and limiting conditions for operation to meet the ALARA criterion. Specific review criteria are contained in SRP Section 11.2.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment related to the liquid waste management systems. The NRC staff concludes that the licensee has adequately accounted for the effects of the increase in fission product and amount of liquid waste on the ability of the liquid waste management systems to control releases of radioactive materials. The NRC staff finds that the liquid waste management systems will continue to meet their design functions following implementation of the proposed EPU. The NRC staff further concludes that the licensee has demonstrated that the liquid waste management systems will continue to meet the requirements of 10 CFR 20.1302, GDCs 60 and 61, and 10 CFR Part 50, Appendix I, Sections II.A and II.D. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the liquid waste management systems.

2.5.5.3 Solid Waste Management Systems

Regulatory Evaluation

The NRC staff's review for the solid waste management systems (SWMS) is focused on the effects that the proposed EPU have on previous analyses and considerations related to the design objectives in terms of expected volumes of waste to be processed and handled, the wet and dry types of waste to be processed, the activity and expected radionuclide distribution contained in the waste, equipment design capacities, and the principal parameters employed in the design of the SWMS. The NRC's acceptance criteria for the SWMS are based on (1) 10 CFR 20.1302 for radioactive materials released in gaseous and liquid effluents to unrestricted areas; (2) GDC-60 for the SWMS being designed with means to handle solid wastes produced during normal plant operation, including operational occurrences; (3) GDCs 63 and 64 for the radioactive waste system being designed for monitoring radiation levels and leakage; and (4) 10 CFR Part 71 for radioactive material packaging. Specific review criteria are contained in SRP Section 11.4.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment related to the SWMS. The NRC staff concludes that the licensee has adequately accounted for the effects of the increase in fission product and amount of solid waste on the ability of the SWMS to process the waste. The NRC staff finds that the SWMS will continue to meet its design functions following implementation of the proposed EPU. The NRC staff further concludes that the licensee has demonstrated that the SWMS will continue to meet the requirements of 10 CFR 20.1302, GDCs 60, 63, and 64, and 10 CFR Part 71. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the SWMS.

2.5.6 Additional Considerations

2.5.6.1. Emergency Diesel Engine Fuel Oil Storage and Transfer System

Regulatory Evaluation

Nuclear power plants are required to have redundant onsite emergency power sources of sufficient capacity to power safety-related equipment (e.g., diesel engine-driven generator sets). This section of the safety evaluation deals with the fuel oil storage and transfer system for these diesel engines. The NRC staff's review focuses on increases in emergency diesel generator electrical demand and the resulting increase in required fuel oil. The NRC's acceptance criteria for the emergency diesel engine fuel oil storage and transfer system are based on (1) GDC-4 for the capability to withstand internally generated missiles, pipe whip, and jet impingement forces associated with pipe breaks; (2) GDC-5 for the capability of shared systems and components important to safety to perform required safety functions; and (3) GDC-17 for the capability of the fuel oil system to meet independence and redundancy criteria. Specific review criteria are contained in SRP Section 9.5.4.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment related to the amount of required fuel oil for the emergency diesel generators and concludes that the licensee has adequately accounted for the effects of the increased electrical demand on fuel oil consumption. The NRC staff concludes that the fuel oil storage and transfer system will continue to provide an adequate amount of fuel oil to allow the diesel generators to meet the onsite power requirements of GDCs 4, 5, and 17. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the fuel oil storage and transfer system.

2.5.6.2 Light Load Handling System (Related to Refueling)

Regulatory Evaluation

The light load handling system (LLHS) includes components and equipment used in handling new fuel at the receiving station to the loading of the spent fuel into the shipping cask. The NRC staff's review covers the avoidance of criticality accidents, radioactivity releases resulting from damage to irradiated fuel, and unacceptable personnel radiation exposures. The NRC staff's review is focused on the effects of the new fuel on system performance and related analyses. The NRC's acceptance criteria for the LLHS are based on (1) GDC-61 for radioactivity release as a result of fuel damage, and the avoidance of excessive personnel radiation exposure; and (2) GDC-62 for criticality accidents. Specific review criteria are contained in SRP Section 9.1.4.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the new fuel on the ability of the LLHS to avoid criticality accidents and concludes that the licensee has adequately incorporated the effects of the new fuel in the analyses. Based on this review, the NRC staff further concludes that the LLHS will continue to meet the requirements of GDCs 61 and 62 for radioactivity releases and prevention of criticality accidents. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the LLHS.

[2.5.7 Additional Review Areas (Plant Systems)]

[Insert Regulatory Evaluation, Technical Evaluation, and Conclusion sections as necessary]

INSERT 6

FOR

SECTION 3.2 - BWR TEMPLATE SAFETY EVALUATION

2.6 Containment Review Considerations

2.6.1 Primary Containment Functional Design

Regulatory Evaluation

The containment encloses the reactor system and is the final barrier against the release of significant amounts of radioactive fission products in the event of an accident. The containment structure must be capable of withstanding, without loss of function, the pressure and temperature conditions resulting from postulated LOCAs, steamline accidents, or feedwater line accidents. The containment structure must continue to serve as a low leakage barrier against the release of fission products for as long as postulated accident conditions require.

The NRC staff's review for the primary containment functional design covers (1) the temperature and pressure conditions in the drywell and wetwell due to a spectrum of postulated LOCAs, (2) the differential pressure across the operating deck for a spectrum of LOCAs (Mark II containments only), (3) suppression pool dynamic effects during a LOCA or following the actuation of one or more RCS safety/relief valves, (4) the consequences of a LOCA occurring within the containment (wetwell), (5) the capability of the containment to withstand the effects of steam bypassing the suppression pool, (6) the suppression pool temperature limit during RCS safety/relief valve operation, and (7) the evaluation of analytical models used for containment analysis. The NRC's acceptance criteria for the primary containment functional design are based on (1) GDC-4 for SSCs important to safety being designed to accommodate the dynamic effects that may occur during normal plant operation or following a LOCA; (2) GDCs 16 and 50 for the containment and its associated systems being able to accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any LOCA; (3) GDC-13 for instrumentation to monitor variables and systems over their anticipated ranges for normal operation and for accident conditions, as appropriate, to assure adequate safety; and (4) GDC-64 for means for monitoring the reactor containment atmosphere for radioactivity that may be released from normal operations and from postulated accidents. Specific review criteria are contained in SRP Section 6.2.1.1.C.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the containment temperature and pressure transient and concludes that the licensee has adequately accounted for the increase of mass and energy resulting from the proposed EPU. The NRC staff further concludes that containment systems will continue to provide sufficient pressure and temperature mitigation capability to ensure that containment integrity is maintained. The NRC staff also concludes that containment systems and instrumentation will continue to be adequate for monitoring containment parameters and release of radioactivity during normal and accident conditions and the containment and associated systems will continue to meet the requirements of GDCs 4, 13, 16, 50, and 64 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to primary containment functional design.

2.6.2 Subcompartment Analyses

Regulatory Evaluation

A subcompartment is defined as any fully or partially enclosed volume within the primary containment that houses high-energy piping and would limit the flow of fluid to the main containment volume in the event of a postulated pipe rupture within the volume. The NRC staff's review for subcompartment analyses covers the determination of the design differential pressure values for containment subcompartments. The NRC staff's review focuses on the effects of the increase in mass and energy release into the containment and the resulting increase in pressurization. The NRC's acceptance criteria for subcompartment analyses are based on (1) GDC-4 for the environmental and missile protection provided to assure that SSCs important to safety are designed to accommodate the dynamic effects that may occur during normal plant operations or during an accident, and (2) GDC-50 for the subcompartments being designed with sufficient margin to prevent fracture of the structure due to pressure differential across the walls of the subcompartment. Specific review criteria are contained in SRP Section 6.2.1.2.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the subcompartment assessment performed by the licensee and the change in predicted pressurization resulting from the increased mass and energy release. The NRC staff concludes that SSCs important to safety will continue to be protected from the dynamic effects resulting from the pipe breaks and that the subcompartments will continue to have sufficient margins to prevent fracture of the structure due to pressure difference across the walls following implementation of the proposed EPU. Based on this review, the NRC staff concludes that the plant will continue to meet GDCs 4 and 50 for the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to subcompartment analyses.

2.6.3 <u>Mass and Energy Release</u>

2.6.3.1 Mass and Energy Release Analysis for Postulated Loss-of-Coolant

<u>Regulatory Evaluation</u>

The release of high energy fluid into containment from pipe breaks could challenge the structural integrity of the containment, including subcompartments and systems within the containment. The NRC staff's review covers the energy sources that are available for release to the containment and the mass and energy release rate calculations for the initial blowdown phase of the accident. The NRC's acceptance criteria for mass and energy release analyses for postulated LOCAs are based on (1) GDC-50 for providing sufficient conservatism in the mass and energy release analysis to assure that containment design margin is maintained and (2) 10 CFR Part 50, Appendix K, for sources of energy during the LOCA. Specific review criteria are contained in SRP Section 6.2.1.3.

<u>Technical Evaluation</u>

[Insert technical evaluation.]

<u>Conclusion</u>

The NRC staff has reviewed the licensee's mass and energy release assessment and concludes that the licensee has adequately addressed the effects of the proposed EPU and appropriately accounts for the sources of energy identified in 10 CFR Part 50, Appendix K. Based on this, the NRC staff finds that the mass and energy release analysis meets the requirements in GDC-50 for ensuring that the analysis is conservative. Therefore, the NRC staff finds the proposed EPU acceptable with respect to mass and energy release for postulated LOCA.

2.6.4 Combustible Gas Control in Containment

Regulatory Evaluation

Following a LOCA, hydrogen and oxygen may accumulate inside the containment due to chemical reaction between the fuel rod cladding and steam, corrosion of aluminum and other materials, and radiolytic decomposition of water. If excessive hydrogen is generated, it may form a combustible mixture in the containment atmosphere. The NRC staff's review covers (1) the production and accumulation of the combustible gases, (2) the capability to prevent high concentrations of combustible gases in local areas, (3) the capability to monitor combustible gas concentrations, and (4) the capability to reduce combustible gas concentrations. The NRC staff's review is primarily focused on any impact that the proposed EPU may have on hydrogen release assumptions, and how increases in hydrogen release are mitigated. The NRC's acceptance criteria for combustible gas control in containment are based on (1) 10 CFR 50.44 and 10 CFR 50.46 for plants being designed to prevent the development of combustible mixtures in the containment atmosphere; (2) GDC-5 for shared systems and components important to safety being able to perform required safety functions; and (3) GDCs 41, 42, and 43 for systems being provided to control the concentration of hydrogen or oxygen that may be released into the reactor containment following postulated accidents to ensure that containment integrity is maintained. ***[Include the following sentence for BWRs with Mark III containments:*** **Additional requirements based on 10 CFR 50.44 for control of combustible gas during severe accidents apply to plants with deliberate ignition systems.]** Specific review criteria are contained in SRP Section 6.2.5.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment related to combustible gas and concludes that the plant will continue to have sufficient capabilities consistent with the requirements in 10 CFR 50.44, 10 CFR 50.46, and GDCs 5, 41, 42, and 43 as discussed in the Regulatory Evaluation section above. Therefore, the NRC staff finds the proposed EPU acceptable with respect to combustible gas control in containment.

2.6.5 Containment Heat Removal

Regulatory Evaluation

Fan cooler systems, spray systems, and residual heat removal (RHR) systems are provided to remove heat from the containment atmosphere and from the water in the containment wetwell. The NRC staff's review in this area focuses on (1) the effects of the proposed EPU on the analyses of the available net positive suction head (NPSH) to the containment heat removal system pumps and (2) the analyses of the heat removal capabilities of the spray water system and the fan cooler heat exchangers. The NRC's acceptance criteria for containment heat removal are based on GDC-38 for the containment heat removal system being capable of rapidly reducing the containment pressure and temperature following a LOCA and maintaining them at acceptably low levels. Specific review criteria are contained in SRP Section 6.2.2, as supplemented by Draft Guide (DG) 1107.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the containment heat removal systems assessment provided by the licensee and concludes that the licensee has adequately addressed the effects of the proposed EPU. The NRC staff finds that the systems will continue to meet GDC-38 for rapidly reducing the containment pressure and temperature following a LOCA and maintaining them at acceptably low levels. Therefore, the NRC staff finds the proposed EPU acceptable with respect to containment heat removal systems.

2.6.6 Secondary Containment Functional Design

Regulatory Evaluation

The secondary containment structure and supporting systems of dual containment plants are provided to collect and process radioactive material that may leak from the primary containment following an accident. The supporting systems maintain a negative pressure within the secondary containment and process this leakage. The NRC staff's review covers (1) analyses of the pressure and temperature response of the secondary containment following accidents within the primary and secondary containments; (2) analyses of the effects of openings in the secondary containment on the capability of the depressurization and filtration system to establish a negative pressure in a prescribed time; (3) analyses of any primary containment leakage paths that bypass the secondary containment; (4) analyses of the pressure response of the secondary containment resulting from inadvertent depressurization of the primary containment when there is vacuum relief from the secondary containment; and (5) the acceptability of the mass and energy release data used in the analysis. The NRC staff's review is primarily focused on the effects that the proposed EPU may have on the pressure and temperature response and drawdown time of the secondary containment, and the impact this may have on offsite dose. The NRC's acceptance criteria for secondary containment functional design are based on (1) GDC-4 for SSCs important to safety being designed to accommodate the effects of environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, and being protected from dynamic effects (e.g., the effects of missiles, pipe whipping, and discharging fluids) that may result from equipment failures; and (2) GDC-16 for reactor containment and associated systems being provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment. Specific review criteria are contained in SRP Section 6.2.3.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment related to the secondary containment pressure and temperature transient and the ability of the secondary containment to provide an essentially leak-tight barrier against uncontrolled release of radioactivity to the environment. The NRC staff concludes that the licensee has adequately accounted for the increase of mass and energy that would result from the proposed EPU and further concludes that the secondary containment and associated systems will continue to provide an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment following implementation of the proposed EPU. Based on this, the NRC staff also concludes that the secondary containment and associated systems will continue to meet the requirements of GDCs 4 and 16. Therefore, the NRC staff finds the proposed EPU acceptable with respect to secondary containment functional design.

[2.6.7 Additional Review Areas (Containment Review Considerations)]

[Insert Regulatory Evaluation, Technical Evaluation, and Conclusion sections as necessary]

INSERT 7

FOR

SECTION 3.2 - BWR TEMPLATE SAFETY EVALUATION

2.7 Habitability, Filtration, and Ventilation

2.7.1 Control Room Habitability System

Regulatory Evaluation

The NRC staff reviews the control room habitability system and control building layout and structures to ensure that plant operators are adequately protected from the effects of accidental releases of toxic and radioactive gases. A further objective of the NRC staff's review is to ensure that the control room can be maintained as the backup center from which technical support center personnel can safely operate the plant in the case of an accident. The NRC staff's review focuses on the effects of the proposed EPU on the radiation doses, toxic gas concentrations, and estimates of dispersion of airborne contamination. The NRC's acceptance criteria for the control room habitability system are based on (1) GDC-4 for accommodating the effects of and being compatible with postulated accidents, including the effects of the release of toxic gases; and (2) GDC-19 for maintaining the control room in a safe, habitable condition during accidents by providing adequate protection against radiation and toxic gases. Specific review criteria are contained in SRP Section 6.4 and other guidance provided in Matrix 7 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment related to the effects of the proposed EPU on the ability of the control room habitability system to protect plant operators against the effects of accidental releases of toxic and radioactive gases. The NRC staff concludes that the licensee has adequately accounted for the increase of toxic and radioactive gases that would result from the proposed EPU and the NRC staff further concludes that the control room habitability system will continue to provide the required protection following implementation of the proposed EPU. Based on this, the NRC staff concludes that the control room habitability system will continue to meet the requirements of GDCs 4 and 19. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the control room habitability system.

2.7.2 Engineered Safety Feature Atmosphere Cleanup

Regulatory Evaluation

ESF atmosphere cleanup systems are designed for fission product removal in postaccident environments. These systems generally include primary systems (e.g., in-containment recirculation) and secondary systems (e.g., standby gas treatment systems and emergency or postaccident air-cleaning systems) for the fuel-handling building, control room, shield building, and areas containing ESF components. For each ESF atmosphere cleanup system, the NRC staff's review focuses on the effects of the proposed EPU on system functional design, environmental design, and provisions to inhibit offdesign temperatures in the adsorber section. The NRC's acceptance criteria for ESF atmosphere cleanup systems are based on (1) GDC-19 for the design of systems for habitability of the control room under accident conditions; (2) GDC-41 for the design of systems for containment atmosphere cleanup following postulated accidents and to control releases to the environment; (3) GDC-61 for the design of systems for radioactivity control under normal and postulated accident conditions; and (4) GDC-64 for monitoring radioactive releases from ESF atmosphere cleanup systems under normal, anticipated operational occurrences, and postulated accident conditions. Specific review criteria are contained in SRP Section 6.5.1.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the ESF atmosphere cleanup systems. The NRC staff concludes that the licensee has adequately accounted for the increase of fission products and changes in expected environmental conditions that would result from the proposed EPU, and the NRC staff further concludes that the ESF atmosphere cleanup systems will continue to provide adequate fission product removal in postaccident environments following implementation of the proposed EPU. Based on this, the NRC staff concludes that the ESF atmosphere cleanup systems will continue to meet the requirements of GDCs 19, 41, 61, and 64. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the ESF atmosphere cleanup systems.

2.7.3 Control Room Area Ventilation System

Regulatory Evaluation

The function of the control room area ventilation system (CRAVS) is to provide a controlled environment for the comfort and safety of control room personnel and to support the operability of control room components during normal operation, anticipated operational occurrences, and DBA conditions. The NRC's review of the CRAVS focuses on the effects that the proposed EPU will have on the functional performance of safety-related portions of the system. The review includes the effects of radiation, combustion, and other toxic products; and the expected environmental conditions in areas served by the CRAVS. The NRC's acceptance criteria for the CRAVS are based on (1) GDC-4 for the CRAVS being designed to accommodate the effects of and to be compatible with anticipated environmental conditions; (2) GDC-19 for providing adequate protection to permit access and occupancy of the control room under accident conditions; and (3) GDC-60 for the system's capability to suitably control release of gaseous radioactive effluents to the environment. Specific review criteria are contained in SRP Section 9.4.1.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the ability of the CRAVS to provide a controlled environment for the comfort and safety of control room personnel and to support the operability of control room components. The NRC staff concludes that the licensee has adequately accounted for the increase of toxic and radioactive gases that would result from the proposed EPU and changes to parameters affecting environmental conditions for control room personnel and equipment. The NRC staff concludes that the CRAVS will continue to provide an acceptable control room environment for safe operation of the plant following implementation of the proposed EPU. The NRC staff also concludes that the system will continue to suitably control the release of gaseous radioactive effluents to the environment. Based on this, the NRC staff concludes that the CRAVS will continue to meet the requirements of GDCs 4, 19, and 60. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the CRAVS.

2.7.4 Spent Fuel Pool Area Ventilation System

Regulatory Evaluation

The function of the spent fuel pool area ventilation system (SFPAVS) is to maintain ventilation in the spent fuel pool equipment areas, permit personnel access, and control airborne radioactivity in the area during normal operation, anticipated operational occurrences, and following postulated fuel handling accidents. The NRC staff's review focuses on the effects of the proposed EPU on the functional performance of the safety-related portions of the system. The NRC's acceptance criteria for the SFPAVS are based on (1) GDC-60 for the system's capability to suitably control release of gaseous radioactive effluents to the environment, and (2) GDC-61 for the system's capability to provide appropriate containment. Specific review criteria are contained in SRP Section 9.4.2.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the SFPAVS. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed EPU on the system's capability to maintain ventilation in the spent fuel pool equipment areas, permit personnel access, control airborne radioactivity in the area, control release of gaseous radioactive effluents to the environment, and provide appropriate containment. Based on this, the NRC staff concludes that the SFPAVS will continue to meet the requirements of GDCs 60 and 61. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the SFPAVS.

2.7.5 Auxiliary and Radwaste Area and Turbine Areas Ventilation Systems

Regulatory Evaluation

The function of the auxiliary and radwaste area ventilation system (ARAVS) and the turbine area ventilation system (TAVS) is to maintain ventilation in the auxiliary and radwaste equipment and turbine areas, permit personnel access, and control the concentration of airborne radioactive material in these areas during normal operation, during anticipated operational occurrences, and after postulated accidents. The NRC staff's review focuses on the effects of the proposed EPU on the functional performance of the safety-related portions of these systems. The NRC's acceptance criteria for the ARAVS and TAVS are based on GDC-60 for the capability of these systems to suitably control release of gaseous radioactive effluents to the environment. Specific review criteria are contained in SRP Sections 9.4.3 and 9.4.4.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the ARAVS and TAVS. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed EPU on the capability of these systems to maintain ventilation in the auxiliary and radwaste equipment areas and in the turbine area, permit personnel access, control the concentration of airborne radioactive material in these areas, and control release of gaseous radioactive effluents to the environment. Based on this, the NRC staff concludes that the ARAVS and TAVS will continue to meet the requirements of GDC-60. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the ARAVS and the TAVS.

2.7.6 Engineered Safety Feature Ventilation System

Regulatory Evaluation

The function of the engineered safety feature ventilation system (ESFVS) is to provide a suitable and controlled environment for ESF components following certain anticipated transients and DBAs. The NRC staff's review for the ESFVS focuses on the effects of the proposed EPU on the functional performance of the safety-related portions of the system. The NRC staff's review also covers (1) the ability of the safety features equipment in the areas being serviced by the ventilation system to function under degraded ESFVS performance; (2) the capability of the ESFVS to circulate sufficient air to prevent accumulation of flammable or explosive gas or fuel-vapor mixtures from components, such as storage batteries and stored fuel; and (3) the capability of the ESFVS to control airborne particulate material (dust) accumulation. The NRC's acceptance criteria for the ESFVS are based on (1) GDC-4 for the ESFVS being designed to accommodate the effects of and to be compatible with anticipated environmental conditions associated with normal operation and postulated accidents; (2) GDC-17 for ensuring proper functioning of the essential electric power system; and (3) GDC-60 for the system being able to suitably control release of gaseous radioactive effluents to the environment. Specific review criteria are contained in SRP Section 9.4.5.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the ESFVS. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed EPU on the ability of the ESFVS to provide a suitable and controlled environment for ESF components. The NRC staff further concludes that the ESFVS will continue to assure a suitable environment for the ESF components following implementation of the proposed EPU. The NRC staff also concludes that the system will continue to suitably control the release of gaseous radioactive effluents to the environment following implementation of the proposed EPU. Based on this, the NRC staff concludes that the ESFVS will continue to meet the requirements of GDCs 4, 17 and 60. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the ESFVS.

[2.7.7 Additional Review Areas (Habitability, Filtration, and Ventilation)]

[Insert Regulatory Evaluation, Technical Evaluation, and Conclusion sections as necessary]

INSERT 8

FOR

SECTION 3.2 - BWR TEMPLATE SAFETY EVALUATION

2.8 Reactor Systems

2.8.1 Fuel System Design

Regulatory Evaluation

The fuel system consists of arrays of fuel rods, burnable poison rods, spacer grids and springs, end plates, channel boxes, and reactivity control rods. The NRC staff reviews the fuel system to ensure that (1) the fuel system is not damaged as a result of normal operation and anticipated operational occurrences, (2) fuel system damage is never so severe as to prevent control rod insertion when it is required, (3) the number of fuel rod failures is not underestimated for postulated accidents, and (4) coolability is always maintained. The NRC staff's review covers fuel system damage mechanisms, limiting values for important parameters, and performance of the fuel system during normal operation, anticipated operational occurrences, and postulated accidents. The NRC's acceptance criteria are based on (1) 10 CFR 50.46 for core cooling; (2) GDC-10 for assuring that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including anticipated operational occurrences; (3) GDC-27 for the reactivity control system being designed with appropriate margin, and in conjunction with the ECCS, being capable of controlling reactivity and cooling the core under postaccident conditions; and (4) GDC-35 for providing an ECCS to transfer heat from the reactor core following any loss of reactor coolant. Specific review criteria are contained in SRP Section 4.2 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed EPU on the fuel system design of the fuel assemblies, control systems, and reactor core. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed EPU on the fuel system and demonstrated that (1) the fuel system will not be damaged as a result of normal operation and anticipated operational occurrences, (2) the fuel system damage will never be so severe as to prevent control rod insertion when it is required, (3) the number of fuel rod failures will not be underestimated for postulated accidents, and (4) coolability will always be maintained. Based on this, the NRC staff concludes that the fuel system and associated analyses will continue to meet the requirements of 10 CFR 50.46, GDC-10, GDC-27, and GDC-35 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the fuel system design.

2.8.2 Nuclear Design

Regulatory Evaluation

The NRC staff reviews the nuclear design of the fuel assemblies, control systems, and reactor core to ensure that fuel design limits will not be exceeded during normal operation or anticipated operational transients, and that the effects of postulated reactivity accidents will not cause significant damage to the RCPB or impair the capability to cool the core. The NRC staff's review covers core power distribution, reactivity coefficients, reactivity control requirements and control provisions, control rod patterns and reactivity worths, criticality, burnup, and vessel irradiation. The NRC's acceptance criteria are based on (1) GDC-10 for assuring that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including anticipated operational occurrences; (2) GDC-11 for the core design to assure that the prompt inherent nuclear feedback characteristics compensate for a rapid increase in reactivity; (3) GDC-12 for precluding or detecting and suppressing power oscillations which could result in conditions exceeding specified acceptable fuel design limits; (4) GDC-13 for instrumentation and controls to monitor variables and systems affecting the fission process over anticipated ranges for normal operation, anticipated operational occurrences and accident conditions, and maintaining the variables and systems within prescribed operating ranges; (5) GDC-20 for automatic initiation of the reactivity control systems to assure that acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and to assure automatic operation of systems and components important to safety under accident conditions; (6) GDC-25 for a single malfunction of the reactivity control system to not cause a violation of the specified acceptable fuel design limits; (7) GDC-26 for providing two independent reactivity control systems of different design, and each system having the capability to control the rate of reactivity changes resulting from planned, normal power changes; (8) GDC-27 for the capability of the reactivity control systems in conjunction with poison addition by the ECCS to reliably control reactivity changes under postulated accident conditions, with appropriate margin for stuck rods; and (9) GDC-28 for the effects of postulated reactivity accidents neither resulting in damage to the RCPB greater than limited local yielding, nor causing sufficient damage to impair significantly the capability to cool the core. Specific review criteria are contained in SRP Section 4.3 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's analyses related to the effect of the proposed EPU on the nuclear design of the fuel assemblies, control systems, and reactor core. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed EPU on the nuclear design and has demonstrated that the fuel design limits will not be exceeded during normal or anticipated operational transients, and that the effects of postulated reactivity accidents will not cause significant damage to the RCPB or impair the capability to cool the core. Based on this evaluation and in coordination with the reviews of the fuel system design, thermal and hydraulic design, and transient and accident analyses, the NRC staff concludes that the nuclear design of the fuel assemblies, control systems, and reactor core will continue to meet the applicable requirements of GDCs 10, 11, 12, 13, 20, 25, 26, 27, and 28. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the nuclear design.

2.8.3 Thermal and Hydraulic Design

Regulatory Evaluation

The NRC staff reviews the thermal and hydraulic design of the core and the RCS to confirm that the design (1) has been accomplished using acceptable analytical methods, (2) is equivalent to or a justified extrapolation from proven designs, (3) provides acceptable margins of safety from conditions which would lead to fuel damage during normal reactor operation and anticipated operational occurrences, and (4) is not susceptible to thermal-hydraulic instability. The review also covers hydraulic loads on the core and RCS components during normal operation and design-basis accident conditions and core thermal-hydraulic stability under normal operation and anticipated transients without scram (ATWS) events. The NRC's acceptance criteria are based on (1) GDC-10 for assuring that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including anticipated operational occurrences; and (2) GDC-12 for the reactor core and associated coolant, control, and protection systems being designed to assure that power oscillations, which can result in conditions exceeding specified acceptable fuel design limits, are not possible or can reliably and readily be detected and suppressed. Specific review criteria are contained in SRP Section 4.4 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed EPU on the thermal and hydraulic design of the core and the RCS. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed EPU on the thermal and hydraulic design and demonstrated that the design has been accomplished using acceptable analytical methods, is **[equivalent to or a justified extrapolation from]** proven designs, provides acceptable margins of safety from conditions which would lead to fuel damage during normal reactor operation and anticipated operational occurrences, and is not susceptible to thermal-hydraulic instability. The NRC staff further concludes that the licensee has adequately accounted for the effects of the proposed EPU on the hydraulic loads on the core and RCS components. Based on this, the NRC staff concludes that the thermal and hydraulic design will continue to meet the requirements of GDCs 10 and 12 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to thermal and hydraulic design.

2.8.4 Emergency Systems

2.8.4.1 Functional Design of Control Rod Drive System

Regulatory Evaluation

The NRC staff's review covers the functional performance of the control rod drive system (CRDS) to confirm that the system can effect a safe shutdown, respond within acceptable limits during anticipated operational occurrences, and prevent or mitigate the consequences of postulated accidents. The review also covers the CRDS cooling system to ensure that it continues to meet its design requirements. The NRC's acceptance criteria are based on (1) GDC-4 for the environmental conditions caused by high or moderate energy pipe breaks during normal plant operation, as well as postulated accidents; (2) GDC-23 for failing into a safe state; (3) GDC-25 for the functional design of redundant reactivity systems to assure that specified acceptable fuel design limits are not exceeded for malfunction of any reactivity control system; (4) GDC-26 for the capability of the reactivity control systems to regulate the rate of reactivity changes resulting from normal operations and anticipated operational occurrences; (5) GDC-27 for the combined capability of reactivity control systems and the emergency core cooling systems to reliably control reactivity changes to assure the capability to cool the core under accident conditions; (6) GDC-28 for postulated reactivity accidents; (7) GDC-29 for functioning under anticipated operational occurrences; and (8) 10 CFR 50.62(c)(3) for diversity of the alternate rod injection system and redundancy of scram air header exhaust valves. Specific review criteria are contained in SRP Section 4.6.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed EPU on the functional design of the CRDS. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed EPU on the system and demonstrated that the system's ability to effect a safe shutdown, respond within acceptable limits, and prevent or mitigate the consequences of postulated accidents will be maintained following the implementation of the proposed EPU. The NRC staff further concludes that the licensee has demonstrated that sufficient cooling exists to ensure the system's design requirements will continue to be met following implementation of the proposed EPU. Based on this, the NRC staff concludes that the fuel system and associated analyses will continue to meet the requirements of GDCs 4, 23, 25, 26, 27, 28, and 29, and 10 CFR 50.62(c)(3) following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the functional design of the CRDS.

2.8.4.2 Overpressure Protection During Power Operation

Regulatory Evaluation

Overpressure protection for the RCPB during power operation is provided by relief and safety valves and the reactor protection system. The NRC staff's review covers relief and safety valves on the main steamlines and piping from these valves to the suppression pool. The NRC's acceptance criteria are based on (1) GDC-15 for the RCS and associated auxiliary, control, and protection systems being designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including anticipated operational occurrences; and (2) GDC-31 for the RCPB being designed with sufficient margin to assure that it behaves in a nonbrittle manner and that the probability of rapidly propagating fracture is minimized. Specific review criteria are contained in SRP Section 5.2.2.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed EPU on the overpressure protection capability of the plant during power operation. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed EPU on pressurization events and overpressure protection features and demonstrated that the plant will continue to have sufficient pressure relief capacity to ensure that pressure limits are not exceeded. Based on this, the NRC staff concludes that the overpressure protection features will continue to provide adequate protection to meet GDCs 15 and 31 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to overpressure protection during power operation.

2.8.4.3 Reactor Core Isolation Cooling System

Regulatory Evaluation

The reactor core isolation cooling (RCIC) system serves as a standby source of cooling water to provide a limited decay heat removal capability whenever the main feedwater system is isolated from the reactor vessel. In addition, the RCIC system may provide decay heat removal necessary for coping with a station blackout. The water supply for the RCIC system comes from the condensate storage tank, with a secondary supply from the suppression pool. The NRC staff's review covers the effect of the proposed EPU on the functional capability of the system. The NRC's acceptance criteria are based on (1) GDC-4 for dynamic effects associated with flow instabilities and loads (e.g., water hammer); (2) GDC-5 for SSCs important to safety not being shared among nuclear power units unless it can be demonstrated that sharing will not impair its ability to perform its safety function; (3) GDC-29 for the system being designed to have an extremely high probability of performing its safety function in the event of anticipated operational occurrences; (4) GDC-33 for the system capability to provide reactor coolant makeup for protection against small breaks in the RCPB so the fuel design limits are not exceeded; (5) GDC-34 for the system design being capable of removing fission product decay heat and other residual heat from the reactor core to preclude fuel damage or RCPB overpressurization; (6) GDC-54 for piping systems penetrating primary containment being provided with leak detection and isolation capabilities; and (7) 10 CFR 50.63 for design provisions to support the plant's ability to withstand and recover from an SBO of a specified duration. Specific review criteria are contained in SRP Section 5.4.6

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed EPU on the ability of the RCIC system to provide decay heat removal following an isolation of main feedwater event and a station blackout event and the ability of the system to provide makeup to the core following a small break in the RCPB. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed EPU on these events and demonstrated that the RCIC system will continue to provide sufficient decay heat removal and makeup for these events following implementation of the proposed EPU. Based on this, the NRC staff concludes that the RCIC system will continue to meet the requirements of GDCs 4, 5, 29, 33, 34 and 54, and 10 CFR 50.63 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the RCIC system.

2.8.4.4 Residual Heat Removal System

Regulatory Evaluation

The RHR system is used to cool down the RCS following shutdown. The RHR system is typically a low pressure system which takes over the shutdown cooling function when the RCS temperature is reduced. The NRC staff's review covers the effect of the proposed EPU on the functional capability of the RHR system to cool the RCS following shutdown and provide decay heat removal. The NRC's acceptance criteria are based on (1) GDC-4 for dynamic effects associated with flow instabilities and loads (e.g., water hammer); (2) GDC-5 which requires that any sharing among nuclear power units of structures, systems, and components important to safety will not significantly impair their safety function; (3) GDC-19 for control room requirements for normal operations and shutdown; and (4) GDC-34 which specifies requirements for an RHR system. Specific review criteria are contained in SRP Section 5.4.7 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed EPU on the RHR system. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed EPU on the system and demonstrated that the RHR system will maintain its ability to cool the RCS following shutdown and provide decay heat removal. Based on this, the NRC staff concludes that the RHR system will continue to meet the requirements of GDCs 4, 5, 19, and 34 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the RHR system.

2.8.4.5 Standby Liquid Control System

Regulatory Evaluation

The standby liquid control system (SLCS) provides backup capability for reactivity control independent of the control rod system. The SLCS functions by injecting a boron solution into the reactor to effect shutdown. The NRC staff's review covers the effect of the proposed EPU on the functional capability of the system to deliver the required amount of boron solution into the reactor. The NRC's acceptance criteria are based on (1) GDC-26 for the requirement that two independent reactivity control systems of different design principles be provided, and the requirement that one of the systems shall be capable of holding the reactor subcritical in the cold condition; (2) GDC-27 for the requirement that the reactivity control systems have a combined capability, in conjunction with poison addition by the ECCS, to reliably control reactivity changes under postulated accident conditions; and (3) 10 CFR 50.62(c)(4) for the SLCS being capable of reliably injecting a borated water solution into the reactor pressure vessel at a boron concentration, boron enrichment, and flow rate that provides sufficient reactivity control and for the system having automatic initiation, where required under the rule, to satisfy ATWS risk-reduction requirements. Specific review criteria are contained in SRP Section 9.3.5 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed EPU on the SLCS and concludes that the licensee has adequately accounted for the effects of the proposed EPU on the system and demonstrated that the system will continue to provide the function of reactivity control independent of the control rod system following implementation of the proposed EPU. Based on this, the NRC staff concludes that the SLCS will continue to meet the requirements of GDCs 26 and 27, and 10 CFR 50.62(c)(4) following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the SLCS.

2.8.5 Accident and Transient Analyses

2.8.5.1 Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Main Steam Relief or Safety Valve

Regulatory Evaluation

Excessive heat removal causes a decrease in moderator temperature which increases core reactivity and can lead to a power level increase and a decrease in shutdown margin. Any unplanned power level increase may result in fuel damage or excessive reactor system pressure. Reactor protection and safety systems are actuated to mitigate the transient. The NRC staff's review covers (1) postulated initial core and reactor conditions, (2) methods of thermal and hydraulic analyses, (3) the sequence of events, (4) assumed reactions of reactor system components, (5) functional and operational characteristics of the reactor protection system, (6) required operator actions, and (7) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC-10 for the RCS being designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operations including anticipated operational occurrences; (2) GDC-15 for the RCS and its associated auxiliaries being designed with appropriate margin to ensure that the RCPB will not be breached during normal operations including anticipated operational occurrences; (3) GDC-20 for the reactor protection system being designed to automatically initiate the operation of appropriate systems, including the reactivity control systems, to ensure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including anticipated operational occurrences; and (4) GDC-26 for the reliable control of reactivity changes to ensure that specified acceptable fuel design limits are not exceeded, including anticipated operational occurrences. Specific review criteria are contained in SRP Section 15.1.1-4 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's analyses of the excess heat removal events described above and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the specified acceptable fuel design limits and the RCPB pressure limits will not be exceeded as a result of these events. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 15, 20, and 26 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the events stated.

2.8.5.2 Decrease in Heat Removal by the Secondary System

2.8.5.2.1. Loss of External Load; Turbine Trip; Loss of Condenser Vacuum; Closure of Main Steam Isolation Valve; and Steam Pressure Regulator Failure (Closed)

Regulatory Evaluation

A number of initiating events may result in unplanned decreases in heat removal by the secondary system. These events result in a sudden reduction in steam flow and, consequently, result in pressurization events. Reactor protection and safety systems are actuated to mitigate the transient. The NRC staff's review covers the sequence of events, the analytical models used for analyses, the values of parameters used in the analytical models, and the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC-10 for the RCS being designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operations, including anticipated operational occurrences; (2) GDC-15 for the RCS and its associated auxiliaries being designed with appropriate margin to ensure that the RCPB will not be breached during normal operations, including anticipated operational occurrences; and (3) GDC-26 for the reliable control of reactivity changes to ensure that specified acceptable fuel design limits are not exceeded during normal operation, including anticipated operational occurrences. Specific review criteria are contained in SRP Section 15.2.1-5 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's analyses of the decrease in heat removal events described above and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the specified acceptable fuel design limits and the RCPB pressure limits will not be exceeded as a result of these events. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 15, and 26 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the events stated.

2.8.5.2.2 Loss of Nonemergency AC Power to the Station Auxiliaries

Regulatory Evaluation

The loss of nonemergency ac power is assumed to result in the loss of all power to the station auxiliaries and the simultaneous tripping of all reactor coolant circulation pumps. This causes a flow coastdown as well as a decrease in heat removal by the secondary system, a turbine trip, an increase in pressure and temperature of the coolant, and a reactor trip. Reactor protection and safety systems are actuated to mitigate the transient. The NRC staff's review covers (1) the sequence of events, (2) the analytical model used for analyses, (3) the values of parameters used in the analytical model, and (4) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC-10 for the RCS being designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operation including anticipated operational occurrences; (2) GDC-15 for the RCS and its associated auxiliaries being designed with appropriate margin to ensure that the RCPB will not be breached during normal operation including anticipated operational occurrences; and (3) GDC-26 for the reliable control of reactivity changes to assure ensure that specified acceptable fuel design limits are not exceeded during normal operation, including anticipated operational occurrences. Specific review criteria are contained in SRP Section 15.2.6 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's analyses of the loss of nonemergency ac power to station auxiliaries event and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the specified acceptable fuel design limits and the RCPB pressure limits will not be exceeded as a result of this event. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 15, and 26 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the loss of nonemergency ac power to station auxiliaries event.

2.8.5.2.3 Loss of Normal Feedwater Flow

Regulatory Evaluation

A loss of normal feedwater flow could occur from pump failures, valve malfunctions, or a LOOP. Loss of feedwater flow results in an increase in reactor coolant temperature and pressure which eventually requires a reactor trip to prevent fuel damage. Decay heat must be transferred from fuel following a loss of normal feedwater flow. Reactor protection and safety systems are actuated to provide this function and mitigate other aspects of the transient. The NRC staff's review covers (1) the sequence of events, (2) the analytical model used for analyses, (3) the values of parameters used in the analytical model, and (4) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC-10 for the RCS being designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operations including anticipated operational occurrences; (2) GDC-15 for the RCS and its associated auxiliaries being designed with appropriate margin to ensure that the RCPB will not be breached during normal operations including anticipated operational occurrences; and (3) GDC-26 for the reliable control of reactivity changes to ensure that specified acceptable fuel design limits are not exceeded during normal operation, including anticipated operational occurrences. Specific review criteria are contained in SRP Section 15.2.7 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's analyses of the loss of normal feedwater flow event and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the specified acceptable fuel design limits and the RCPB pressure limits will not be exceeded as a result of the loss of normal feedwater flow. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 15, and 26 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the loss of normal feedwater flow event.

2.8.5.3 Decrease in Reactor Coolant System Flow

2.8.5.3.1 Loss of Forced Reactor Coolant Flow

Regulatory Evaluation

A decrease in reactor coolant flow occurring while the plant is at power could result in a degradation of core heat transfer. An increase in fuel temperature and accompanying fuel damage could then result if specified acceptable fuel design limits are exceeded during the transient. Reactor protection and safety systems are actuated to mitigate the transient. The NRC staff's review covers (1) the postulated initial core and reactor conditions, (2) the methods of thermal and hydraulic analyses, (3) the sequence of events, (4) assumed reactions of reactor systems components, (5) the functional and operational characteristics of the reactor protection system, (6) required operator actions, and (7) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC-10 for the RCS being designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operations, including anticipated operational occurrences; (2) GDC-15 for the RCS and its associated auxiliaries being designed with appropriate margin to ensure that the RCPB will not be breached during normal operations, including anticipated operational occurrences; and (3) GDC-26 for the reliable control of reactivity changes to ensure that specified acceptable fuel design limits are not exceeded during normal operation, including anticipated operational occurrences. Specific review criteria are contained in SRP Section 15.3.1-2 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's analyses of the decrease in reactor coolant flow event and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the specified acceptable fuel design limits and the RCPB pressure limits will not be exceeded as a result of this event. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 15, and 26 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the decrease in reactor coolant flow event.

2.8.5.3.2 Reactor Recirculation Pump Rotor Seizure and Reactor Recirculation Pump Shaft Break

Regulatory Evaluation

The events postulated are an instantaneous seizure of the rotor or break of the shaft of a reactor recirculation pump. Flow through the affected loop is rapidly reduced, leading to a reactor and turbine trip. The sudden decrease in core coolant flow while the reactor is at power results in a degradation of core heat transfer which could result in fuel damage. The initial rate of reduction of coolant flow is greater for the rotor seizure event. However, the shaft break event permits a greater reverse flow through the affected loop later during the transient and, therefore, results in a lower core flow rate at that time. In either case, reactor protection and safety systems are actuated to mitigate the transient. The NRC staff's review covers (1) the postulated initial and long-term core and reactor conditions, (2) the methods of thermal and hydraulic analyses, (3) the sequence of events, (4) the assumed reactions of reactor system components, (5) the functional and operational characteristics of the reactor protection system, (6) required operator actions, and (7) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC-27 and GDC-28 for the RCS being designed with appropriate margin to ensure that the capability to cool the core is maintained; and (2) GDC-31 for the RCS being designed with sufficient margin to ensure that the RCPB behaves in a nonbrittle manner and that the probability of propagating fracture is minimized. Specific review criteria are contained in SRP Section 15.3.3-4 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's analyses of the sudden decrease in core coolant flow events and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the ability to insert control rods is maintained, the RCPB pressure limits will not be exceeded, the RCPB will behave in a nonbrittle manner, the probability of propagating fracture of the RCPB is minimized, and adequate core cooling will be provided. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 27, 28, and 31 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the sudden decrease in core coolant flow events.

2.8.5.4 Reactivity and Power Distribution Anomalies

2.8.5.4.1. Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition

Regulatory Evaluation

An uncontrolled control rod assembly withdrawal from subcritical or low power startup condition may be caused by a malfunction of the reactor control or rod control systems. This withdrawal will uncontrollably add positive reactivity to the reactor core, resulting in a power excursion. The NRC staff's review covers (1) the description of the causes of the transient and the transient itself, (2) the initial conditions, (3) the reactor parameters used in the analysis, (4) the analytical methods and computer codes used, and (5) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC-10 for the RCS being designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operations, including anticipated operational occurrences; (2) GDC-20 for the reactor protection system being designed to initiate automatically the operation of appropriate systems, including the reactivity control systems, to ensure that specified acceptable fuel design limits are not exceeded during normal operation, including anticipated operational occurrences; and (3) GDC-25 for the functional design of redundant reactivity control systems to assure that specified acceptable fuel design limits are not exceeded in the event of a single malfunction of the reactivity control systems. Specific review criteria are contained in SRP Section 15.4.1 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's analyses of the uncontrolled control rod assembly withdrawal from a subcritical or low power startup condition and concludes that the licensee's analyses have adequately accounted for the changes in core design required for operation of the plant at the proposed power level. The NRC staff also concludes that the licensee's analyses were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure the specified acceptable fuel design limits are not exceeded. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 20, and 25 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the uncontrolled control rod assembly withdrawal from a subcritical or low power startup condition.

2.8.5.4.2 Uncontrolled Control Rod Assembly Withdrawal at Power

Regulatory Evaluation

An uncontrolled control rod assembly withdrawal at power may be caused by a malfunction of the reactor control or rod control systems. This withdrawal will uncontrollably add positive reactivity to the reactor core, resulting in a power excursion. The NRC staff's review covers (1) the description of the causes of the anticipated operational occurrence and the description of the event itself, (2) the initial conditions, (3) the reactor parameters used in the analysis, (4) the analytical methods and computer codes used, and (5) the results of the associated analyses. The NRC's acceptance criteria are based on (1) GDC-10 for the RCS being designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operations, including anticipated operational occurrences; (2) GDC-20 for the reactor protection system being designed to initiate automatically the operation of appropriate systems, including the reactivity control systems, to ensure that specified acceptable fuel design limits are not exceeded during normal operation, including anticipated operational occurrences; and (3) GDC-25 for the functional design of redundant reactivity control systems to assure that specified acceptable fuel design limits are not exceeded in the event of a single malfunction of the reactivity control systems. Specific review criteria are contained in SRP Section 15.4.2 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's analyses of the uncontrolled control rod assembly withdrawal at power event and concludes that the licensee's analyses have adequately accounted for the changes in core design required for operation of the plant at the proposed power level. The NRC staff also concludes that the licensee's analyses were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure the specified acceptable fuel design limits are not exceeded. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 20, and 25 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the uncontrolled control rod assembly withdrawal at power.

2.8.5.4.3 Startup of a Recirculation Loop at an Incorrect Temperature and Flow Controller Malfunction Causing an Increase in Core Flow Rate

<u>Regulatory Evaluation</u>

A startup of an inactive loop transient may result in either an increased core flow or the introduction of cooler water into the core. This event causes an increase in core reactivity due to decreased moderator temperature and core void fraction. The NRC staff's review covers (1) the sequence of events, (2) the analytical model, (3) the values of parameters used in the analytical model, and (4) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC-10 and GDC-20 for the RCS being designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operations, including anticipated operational occurrences; (2) GDC-15 and GDC-28 for the RCS and its associated auxiliaries being designed with appropriate margin to ensure that the RCPB will not be breached during normal operations, including anticipated operational occurrences; and (3) GDC-26 for the reliable control of reactivity changes to ensure that specified acceptable fuel design limits are not exceeded during normal operations, including anticipated operational occurrences. Specific review criteria are contained in SRP Section 15.4.4-5 and other guidance provided in Matrix 8 of RS-001.

<u>Technical Evaluation</u>

[Insert technical evaluation.]

<u>Conclusion</u>

The NRC staff has reviewed the licensee's analyses of the increase in core flow event and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the specified acceptable fuel design limits and the RCPB pressure limits will not be exceeded as a result of this event. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 15, 20, 26, and 28 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the increase in core flow event.

2.8.5.4.4 Spectrum of Rod Drop Accidents

Regulatory Evaluation

The NRC staff evaluates the consequences of a control rod drop accident in the area of physics. The NRC staff's review covers the occurrences that lead to the accident, safety features designed to limit the amount of reactivity available and the rate at which reactivity can be added to the core, the analytical model used for analyses, and the results of the analyses. The NRC's acceptance criteria are based on GDC-28 for the effects of postulated reactivity accidents, neither resulting in damage to the RCPB greater than limited local yielding nor causing sufficient damage to impair significantly the capacity to cool the core. Specific review criteria are contained in SRP Section 15.4.9 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's analyses of the rod drop accident and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that appropriate reactor protection and safety systems will prevent postulated reactivity accidents that could (1) result in damage to the RCPB greater than limited local yielding, or (2) cause sufficient damage that would significantly impair the capability to cool the core. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDC-28 following implementation of the EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the rod drop accident.

2.8.5.5 Inadvertent Operation of ECCS or Malfunction that Increases Reactor Coolant Inventory

Regulatory Evaluation

Equipment malfunctions, operator errors, and abnormal occurrences could cause unplanned increases in reactor coolant inventory. Depending on the temperature of the injected water and the response of the automatic control systems, a power level increase may result and, without adequate controls, could lead to fuel damage or overpressurization of the RCS. Alternatively, a power level decrease or depressurization may result. Reactor protection and safety systems are actuated to mitigate these events. The NRC staff's review covers (1) the sequence of events, (2) the analytical model used for analyses, (3) the values of parameters used in the analytical model, and (4) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC-10 for the RCS being designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operations, including anticipated operational occurrences; (2) GDC-15 for the RCS and its associated auxiliaries being designed with appropriate margin to ensure that the RCPB will not be breached during normal operations, including anticipated operational occurrences; and (3) GDC-26 for the reliable control of reactivity changes to ensure that specified acceptable fuel design limits are not exceeded during normal operation, including anticipated operational occurrences. Specific review criteria are contained in SRP Section 15.5.1-2 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's analyses of the inadvertent operation of ECCS or malfunction that increases reactor coolant inventory and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the specified acceptable fuel design limits and the RCPB pressure limits will not be exceeded as a result of this event. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 15, and 26 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the inadvertent operation of ECCS or malfunction that increases reactor coolant inventory.

2.8.5.6 Decrease in Reactor Coolant Inventory

2.8.5.6.1 Inadvertent Opening of a Pressure Relief Valve

Regulatory Evaluation

The inadvertent opening of a pressure relief valve results in a reactor coolant inventory decrease and a decrease in RCS pressure. The pressure relief valve discharges into the suppression pool. Normally there is no reactor trip. The pressure regulator senses the RCS pressure decrease and partially closes the turbine control valves (TCVs) to stabilize the reactor at a lower pressure. The reactor power settles out at nearly the initial power level. The coolant inventory is maintained by the feedwater control system using water from the condensate storage tank via the condenser hotwell. The NRC staff's review covers (1) the sequence of events, (2) the analytical model used for analyses, (3) the values of parameters used in the analytical model, and (4) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC-10 for the RCS being designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operations, including anticipated operational occurrences; (2) GDC-15 for the RCS and its associated auxiliaries being designed with appropriate margin to ensure that the RCPB will not be breached during normal operations, including anticipated operational occurrences; and (3) GDC-26 for the reactivity control systems to provide adequate control of reactivity changes to ensure that the acceptable fuel design limits are not exceeded during normal operations and anticipated transients during normal operations, inlcuding anticipated operational occurrences. Specific review criteria are contained in SRP Section 15.6.1 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's analyses of the inadvertent opening of a pressure relief valve event and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the specified acceptable fuel design limits and the RCPB pressure limits will not be exceeded as a result of this event. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 15, and 26 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the inadvertent opening of a pressure relief valve event.

2.8.5.6.2 Emergency Core Cooling System and Loss-of-Coolant Accidents

Regulatory Evaluation

LOCAs are postulated accidents that would result from the loss of reactor coolant from piping breaks in the RCPB at a rate in excess of the capability of the normal reactor coolant makeup system. Loss of significant quantities of reactor coolant would prevent heat removal from the reactor core, unless the water is replenished. The reactor protection and ECCS systems are provided to mitigate this accidents. The NRC staff's review covers (1) the licensee's determination of break locations and break sizes; (2) postulated initial conditions; (3) the sequence of events; (4) the analytical model used for analyses, calculations of the reactor power, pressure, flow, and temperature transients; (5) calculations for peak cladding temperature, total oxidation of the cladding, total hydrogen generation, changes in core geometry, and long-term cooling; (6) functional and operational characteristics of the reactor protective and ECCS systems; and (7) required operator actions. The NRC's acceptance criteria are based on (1) 10 CFR 50.46 and Appendix K to 10 CFR Part 50 for the use of an acceptable evaluation model for LOCA analyses and ECCS equipment being provided that refills the vessel in a timely manner for a LOCA; (2) GDC-4 for the dynamic effects associated with flow instabilities and loads (e.g., water hammer); (3) GDC-27 for the ECCS design having the capability to assure that under postulated accident conditions and with appropriate margin for stuck rods, the capability to cool the core is maintained; and (4) GDC-35 for the ECCS being designed to provide an abundance of core cooling to transfer heat from the core at a rate so that fuel and clad damage will not interfere with continued effective core cooling. Specific review criteria are contained in SRP Sections 6.3 and 15.6.5 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's analyses of the LOCA events and the ECCS. The NRC staff concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and that the analyses were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection system and the ECCS will continue to ensure that the peak cladding temperature, total oxidation of the cladding, total hydrogen generation, and changes in core geometry, and long-term cooling will remain within acceptable limits. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 4, 27, 35, and 10 CFR 50.46 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the LOCA.

2.8.5.7 Anticipated Transients Without Scrams

Regulatory Evaluation

ATWS is defined as an anticipated operational occurrence followed by the failure of the reactor portion of the protection system specified in GDC-20. The regulation at 10 CFR 50.62 requires that:

- each BWR have an alternate rod injection (ARI) system that is designed to perform its function in a reliable manner and be independent (from the existing reactor trip system) from sensor output to the final actuation device.

- each BWR have a standby liquid control system (SLCS) with the capability of injecting a borated water solution with reactivity control equivalent to the control obtained by injecting 86 gpm of a 13 weight-percent sodium pentaborate decahydrate solution at the natural boron-10 isotope abundance into a 251-inch inside diameter reactor vessel. The system initiation must be automatic.

- each BWR have equipment to trip the reactor coolant recirculation pumps automatically under conditions indicative of an ATWS.

The NRC staff's review is conducted to ensure that (1) the above requirements are met, (2) sufficient margin is available in the setpoint for the SLCS pump discharge relief valve such that SLCS operability is not affected, and (3) operator actions specified in the plant's Emergency Operating Procedures are consistent with the generic emergency procedure guidelines/severe accident guidelines (EPGs/SAGs). In addition, the NRC staff reviews the licensee's ATWS analysis to ensure that (1) the peak vessel bottom pressure is less than ASME Service Level C limit of 1500 psig; (2) the peak clad temperature is within the 10 CFR 50.46 limit of 2200 °F; (3) the peak suppression pool temperature is less than the design limit; and (4) the peak containment pressure is less than the containment design pressure. The NRC staff also evaluates the potential for thermal-hydraulic instability in conjunction with ATWS events using the methods and criteria approved by the NRC staff. For this analysis, the NRC staff reviews the limiting event determination, the sequence of events, the analytical model and its applicability, the values of parameters used in the analytical model, and the results of the analyses. If the licensee relies upon generic vendor analyses, the NRC staff reviews the licensee's justification of the applicability of that analysis to its plant and the operating conditions for the proposed EPU. Review guidance is provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the information submitted by the licensee related to ATWS and concludes that the licensee has adequately accounted for the effects of the proposed EPU on ATWS. The NRC staff concludes that the licensee has demonstrated that ARI, SLCS, and recirculation pump trip systems have been installed and that they will continue to meet the requirements of 10 CFR 50.62 and the analysis acceptance criteria following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to ATWS.

2.8.6 Fuel Storage

2.8.6.1 New Fuel Storage

Regulatory Evaluation

Nuclear reactor plants include facilities for the storage of new fuel. The quantity of new fuel to be stored varies from plant to plant, depending upon the specific design of the plant and the individual refueling requirements. The NRC staff's review covers the ability of the storage facilities to maintain the new fuel in a subcritical array during all credible storage conditions. The review focuses on the effect of changes in fuel design on the analyses for the new fuel storage facilities. The NRC's acceptance criteria are based on GDC-62 for the prevention of criticality by physical systems or processes utilizing geometrically safe configurations. Specific review criteria are contained in SRP Section 9.1.1.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's analyses related to the effect of the new fuel on the analyses for the new fuel storage facilities and concludes that the new fuel storage facilities will continue to meet the requirements of GDC-62 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the new fuel storage.

2.8.6.2 Spent Fuel Storage

Regulatory Evaluation

Nuclear reactor plants include storage facilities for the wet storage of spent fuel assemblies. The safety function of the spent fuel pool and storage racks is to maintain the spent fuel assemblies in a safe and subcritical array during all credible storage conditions and to provide a safe means of loading the assemblies into shipping casks. The NRC staff's review covers the effect of the proposed EPU on the criticality analysis (e.g., reactivity of the spent fuel storage array and boraflex degradation or neutron poison efficacy). The NRC's acceptance criteria are based on (1) GDC-4 for the facility being capable to withstand the effects of environmental conditions such that safety functions will not be precluded and (2) GDC-62 for the prevention of criticality by physical systems or processes utilizing geometrically safe configurations. Specific review criteria are contained in SRP Section 9.1.2.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed EPU on the spent fuel storage capability and concludes that the licensee has adequately accounted for the effects of the proposed EPU on the spent fuel rack temperature and criticality analyses. The NRC staff also concludes that the spent fuel pool design will continue to ensure an acceptably low temperature and an acceptable degree of subcriticality following implementation of the proposed EPU. Based on this, the NRC staff concludes that the spent fuel storage facilities will continue to meet the requirements of GDCs 4 and 62 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to spent fuel storage.

[2.8.7 Additional Review Areas (Reactor Systems)]

[Insert Regulatory Evaluation, Technical Evaluation, and Conclusion sections as necessary]

INSERT 9

FOR

SECTION 3.2 - BWR TEMPLATE SAFETY EVALUATION

2.9 Source Terms and Radiological Consequences Analyses

2.9.1 Source Terms for Radwaste Systems Analyses

Regulatory Evaluation

The NRC staff reviews the radioactive source term associated with EPUs to ensure the adequacy of the sources of radioactivity used by the licensee as input to verify that the radioactive waste management systems have adequate capacity for the treatment of radioactive liquid and gaseous wastes. The NRC staff's review includes the parameters used to determine (1) the concentration of each radionuclide in the reactor coolant, (2) the fraction of fission product activity released to the reactor coolant, (3) concentrations of all nonfission product radionuclides in the reactor coolant, (4) leakage rates and associated fluid activity of all potentially radioactive water and steam systems, and (5) potential sources of radioactive materials in effluents that are not considered in the plant's [**Updated Safety Analysis Report** ***or*** **Updated Final Safety Analysis Report**] related to liquid waste management systems and gaseous waste management systems. The NRC's acceptance criteria for source terms are based on (1) 10 CFR Part 20 for radioactivity in liquid and gaseous effluents released to unrestricted areas; (2) 10 CFR Part 50, Appendix I, for the numerical guides for design objectives and limiting conditions for operation to meet the "as low as reasonably achievable" criterion; and (3) GDC-60 for the radioactive waste management systems being able to control the releases of radioactive liquid and gaseous effluents to the environment. Specific review criteria are contained in SRP Section 11.1.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the radioactive source term associated with the proposed EPU and concludes that the proposed parameters and resultant composition and quantity of radionuclides are appropriate for the evaluation of the radioactive waste management systems. The NRC staff further concludes that the proposed radioactive source term meets the requirements of 10 CFR Part 20, 10 CFR Part 50, Appendix I, and GDC-60. Therefore, the NRC staff finds the proposed EPU acceptable with respect to source terms.

NOTE: Use Sections 2.9.2 and 2.9.3 below if the licensee's radiological consequences analyses are based on an alternative source term.

2.9.2 Radiological Consequences Analyses Using Alternative Source Terms

NOTE: There are two cases that may be encountered here: (1) a licensee may be implementing an alternative source term for the first time, or (2) a licensee may have already fully implemented an alternative source term and is revising the previously approved dose analyses that use alternative source term methodologies. The second paragraph for each heading is only needed for a first-time implementation of an alternative source term (either partial or full implementations). Several accidents may have been analyzed - see corresponding SRP sections for further regulatory evaluation text (to be modified), as needed.

Regulatory Evaluation

The NRC staff reviews the DBA radiological consequences analyses. The radiological consequences analyses reviewed are the LOCA, fuel handling accident (FHA), control rod drop accident (CRDA), and main steamline break (MSLB). The NRC staff's review for each accident analysis includes (1) the sequence of events; and (2) models, assumptions, and parameter inputs used by the licensee for the calculation of the total effective dose equivalent. The NRC's acceptance criteria for radiological consequences analyses using an alternative source term are based on (1) GDC-19 for control room habitability and (2) 10 CFR 50.67 for radiological consequences of a postulated accident. Specific review criteria are contained in SRP Section 15.0.1.

NOTE: Use the following paragraph for a first implementation of an alternative source term:

The NRC staff reviews the implementation of alternative source terms. The NRC's acceptance criteria for implementation of alternative source terms are based on (1) 10 CFR 50.49 for qualification of safety-related equipment with regard to integrated radiation dose during normal and accident conditions; (2) 10 CFR 50.67 for the implementation of an alternative source term in current operating nuclear power plants; (3) GDC-19 for maintaining the control room in a safe, habitable condition under accident conditions by providing adequate protection against radiation and toxic gases; (4) 10 CFR Part 51 for environmental assessments of radioactive material releases during normal and accident conditions; (5) Paragraph IV.E.8 of 10 CFR Part 50, Appendix E, for maintaining emergency facilities in a safe, habitable condition under accident conditions by providing adequate protection against radiation and toxic gases; and (6) plant-specific licensing commitments made in response to the NUREG-0737 (Items II.B.2, II.B.3, II.F.1, III.D.1.1, III.A.1.2, and III.D.3.4). Specific review criteria are contained in SRP Sections 15.0.1.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has evaluated the licensee's revised accident analyses performed in support of the proposed EPU and concludes that the licensee has adequately accounted for the effects of the proposed EPU. The NRC staff further concludes that the plant site and the dose-mitigating engineered safety features (ESFs) remain acceptable with respect to the radiological consequences of postulated DBAs since the calculated total effective dose equivalent (TEDE) at the exclusion area boundary (EAB), at the low population zone (LPZ) outer boundary, and in the control room meet the exposure guideline values specified in 10 CFR 50.67 and GDC-19, as well as applicable acceptance criteria denoted in SRP Section 15.0.1. Therefore, the NRC staff finds the licensee's proposed EPU acceptable with respect to the radiological consequences of DBAs.

NOTE: Use the following paragraph for a first implementation of an alternative source term:

The NRC staff has reviewed the alternative source term methodology used by the licensee in evaluating the effects of the proposed EPU and concludes that changes continue to provide sufficient margin of safety with adequate defense-in-depth to address unanticipated events and to compensate for uncertainties in accident progression, analysis assumptions, and parameter inputs. Therefore, the NRC staff finds the licensee's proposed EPU acceptable with respect to the implementation of an alternative source term.

[2.9.3 Additional Review Areas (Radiological Consequences Analyses)]

[Insert Regulatory Evaluation, Technical Evaluation, and Conclusion sections as necessary]

NOTE: Use Sections 2.9.2 - 2.9.8 below if the licensee's radiological consequences analyses are not based on an alternative source term (i.e., if the analyses are based on a traditional source term (i.e., TID-14844)

2.9.2 Radiological Consequences of Control Rod Drop Accident

Regulatory Evaluation

The NRC staff reviews the analyses of the radiological consequences of a control rod drop accident (CRDA). The NRC staff's review includes an examination of (1) the plant's response to the accident, (2) the release of fission products from the core to the environment via the turbine and condensers as a result of the accident, (3) and the calculation of radiological doses at the exclusion area boundary (EAB) and low population zone (LPZ) outer boundary, and in the control room due to the releases from the accident. The NRC's acceptance criteria for the radiological consequences of a control rod drop accident are based on (1) GDC-19 for control room habitability and (2) 10 CFR Part 100 for mitigating the radiological consequences of an accident. Specific review criteria are contained in SRP Sections 6.4 and 15.4.9.A, and other guidance provided in Matrix 9 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has evaluated the licensee's revised accident analyses for the radiological consequences of a control rod drop accident and concludes that the licensee has adequately accounted for the effects of the proposed EPU on these analyses. The NRC staff further concludes that the plant site and the dose-mitigating ESFs remain acceptable with respect to the radiological consequences of a postulated control rod drop accident since the calculated whole-body and thyroid doses at the EAB and the LPZ outer boundary are well within the exposure guideline values in 10 CFR 100.11. The NRC staff also concludes that the control room meets the dose requirements of GDC-19 for DBAs. Therefore, the NRC staff finds the licensee's proposed EPU acceptable with respect to the radiological consequences of a control rod drop accident.

2.9.3 Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment

Regulatory Evaluation

The NRC staff reviews the analysis of the radiological consequences of failures outside the containment of small lines connected to the primary coolant pressure boundary (e.g., instrument lines and sample lines). The NRC staff's review includes (1) the identification of small lines postulated to fail and the isolation provisions for these lines; (2) the failure scenario; (3) the models and assumptions for the calculation of the radiological doses for the postulated failure; and (4) an evaluation of the primary coolant iodine activity, including the effects of a concurrent iodine spike, and the TSs for the reactor coolant iodine activity. The NRC's acceptance criteria for the radiological consequences of failures outside the containment of small lines connected to the primary coolant pressure boundary are based on (1) GDC-19 for control room habitability and (2) GDC-55 for the isolation requirements of small-diameter lines connected to the primary system that are acceptable on the basis of meeting 10 CFR 100.11. Specific review criteria are contained in SRP Sections 6.4 and 15.6.2, and other guidance provided in Matrix 9 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has evaluated the licensee's revised accident analyses for the radiological consequences of failures outside the containment of small lines connected to the primary coolant pressure boundary and concludes that the licensee has adequately accounted for the effects of the proposed EPU on these analyses. The NRC staff further concludes that the plant site and the dose-mitigating ESFs remain acceptable with respect to the radiological consequences of a postulated failure outside the containment of a small line carrying reactor coolant since the calculated whole-body and thyroid doses at the EAB and the LPZ outer boundary are substantially below the exposure guideline values of 10 CFR 100.11. The NRC staff also concludes that the control room meets the dose requirements of GDC-19 for DBAs. Therefore, the NRC staff finds the licensee's proposed EPU acceptable with respect to the radiological consequences of failures outside the containment of small lines connected to the primary coolant pressure boundary.

2.9.4 Radiological Consequences of Main Steamline Failure Outside Containment

Regulatory Evaluation

The NRC staff reviews the analyses of the radiological consequences of an MSLB accident outside the containment to ensure that radioactive releases due to the failure are adequately limited by the TS limit on primary coolant activity. The NRC staff's review includes two cases for the reactor coolant iodine concentration: (1) with a preaccident iodine spike and (2) with the maximum equilibrium concentration for continued full-power operation. The NRC's acceptance criteria for the radiological consequences of an MSLB outside containment are based on (1) GDC-19 for control room habitability and (2) 10 CFR Part 100 for the radiological consequences of an accident. Specific review criteria are contained in SRP Sections 6.4 and 15.6.4, and other guidance provided in Matrix 9 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has evaluated the licensee's revised accident analyses for the radiological consequences of an MSLB outside containment and concludes that the licensee has adequately accounted for the effects of the proposed EPU on the analyses. The NRC staff further concludes that the plant site and the dose-mitigating ESFs remain acceptable with respect to the radiological consequences of a postulated MSLB outside containment since the calculated whole-body and thyroid doses at the EAB and the LPZ outer boundary do not exceed the exposure guideline values of 10 CFR 100.11 (assuming a preaccident iodine spike) and are a small fraction of the Part 100 values for an MSLB with the primary coolant at the maximum equilibrium concentration for continued full-power operation. The NRC staff also concludes that the control room meets the dose requirements of GDC-19 for DBAs. Therefore, the NRC staff finds the licensee's proposed EPU acceptable with respect to a postulated failure of an MSLB outside containment.

2.9.5 Radiological Consequences of a Design-Basis Loss-of-Coolant Accident

Regulatory Evaluation

The NRC staff reviews the analyses of the radiological consequences of a design-basis LOCA. This review includes a summary review of the doses from the hypothetical design-basis LOCA and a specific review of the doses from containment leakage and leakage from ESF components outside containment that contribute to the total LOCA doses. The NRC staff's review also includes (1) the contribution to the dose due to leakage from the main steam isolation valves (MSIVs); (2) the methodology and results of calculations of the radiological consequences resulting from containment and ESF components and MSIV leakage following a hypothetical LOCA; and (3) an assessment of the containment with respect to the assumptions and the input parameters for the dose calculations. The NRC's calculations are based on pertinent information in the [**Updated Safety Analysis Report *or* Updated Final Safety Analysis Report**] and considers the NRC staff's evaluation of dose-mitigating ESFs. The NRC's acceptance criteria for the radiological consequences of a design-basis LOCA are based on (1) GDC-19 for control room habitability and (2) 10 CFR Part 100 for mitigating the radiological consequences of an accident. Specific review criteria are contained in SRP Section 6.4 and Appendices A, B, and D of SRP Section 15.6.5, and other guidance provided in Matrix 9 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has evaluated the licensee's revised accident analyses for the radiological consequences of a design-basis LOCA and concludes that the licensee has adequately accounted for the effects of the proposed EPU on the analyses. The NRC staff further concludes that the plant site and the dose-mitigating ESFs remain acceptable with respect to the radiological consequences of a design-basis LOCA since the calculated whole-body and thyroid doses at the EAB and the LPZ outer boundary do not exceed the exposure guideline values of 10 CFR 100.11 and the calculated doses in the control room meet the requirements of GDC-19. Therefore, the NRC staff finds the licensee's proposed EPU acceptable with respect to the radiological consequences of a design-basis LOCA.

2.9.6 Radiological Consequences of Fuel Handling Accidents

Regulatory Evaluation

The NRC staff reviews the analyses of the radiological consequences of a postulated FHA. The purpose of this review is to evaluate the adequacy of system design features and plant procedures provided for the mitigation of the radiological consequences of accidents that involve damage to spent fuel. Such accidents include the dropping of a single fuel assembly and handling tool or a heavy object onto other spent fuel assemblies. Such accidents may occur inside the containment, along the fuel transfer canal, and in the fuel building. The NRC staff's review includes (1) the sequence of events, models, and assumptions used by the licensee for the calculation of the radiological doses; (2) the adequacy of the ESFs provided for the purpose of mitigating potential accident doses; and (3) the containment ventilation system with respect to its function as a dose-mitigating ESF system, including the radiation detection system on the containment purge/vent lines for those plants that will vent or purge the containment during fuel handling operations. The NRC's acceptance criteria for the radiological consequences of FHAs are based on (1) GDC-19 for control room habitability; (2) GDC-61 for appropriate containment, confinement, and filtering systems; and (3) 10 CFR Part 100 for the calculated radiological consequences of FHAs. Specific review criteria are contained in SRP Sections 6.4 and 15.7.4, and other guidance provided in Matrix 9 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has evaluated the licensee's revised accident analyses for the radiological consequences of FHAs and concludes that the licensee has adequately accounted for the effects of the proposed EPU on these analyses. The NRC staff further concludes that the plant site and the dose-mitigating ESFs remain acceptable with respect to the radiological consequences of a postulated FHA since the calculated whole-body and thyroid doses at the EAB and the LPZ outer boundary are well within the exposure guideline values of 10 CFR 100.11 and GDC-61. The NRC staff also concludes that the control room meets the dose requirements of GDC-19 for DBAs. Therefore, the NRC staff finds the licensee's proposed EPU acceptable with respect to the radiological consequences of FHAs.

2.9.7 Radiological Consequences of Spent Fuel Cask Drop Accidents

Regulatory Evaluation

The NRC staff reviews the analyses of the radiological consequences of the release of fission products from irradiated fuel in a spent fuel cask that is postulated to drop during cask handling operations. The NRC staff's review is conducted to verify various design and operation aspects of the system. The NRC staff's review includes (1) determining a need for a design-basis radiological analysis sequence of events; (2) models and assumptions used by the licensee for the calculation of the radiological doses; (3) comparing calculated doses to exposure guidelines to determine the acceptability of the EAB and LPZ outer boundary distances and to confirm the adequacy of ESFs provided for the purpose of mitigating potential doses from spent fuel cask drop accidents, including the effects on control room habitability; and (4) examining the relationship of the operational modes of the standby gas treatment system (SGTS) to the time sequence of the accident in order to give proper credit, in a dual containment design where the fuel building may be exhausted through the SGTS. The NRC's acceptance criteria for the radiological consequences of spent fuel cask drop accidents are based on (1) GDC-19 for control room habitability; (2) GDC-61 for appropriate containment, confinement and filtering systems; and (3) 10 CFR Part 100 for the calculated radiological consequences of a spent fuel cask drop accident. Specific review criteria are contained in SRP Sections 6.4 and 15.7.5, and other guidance provided in Matrix 9 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has evaluated the licensee's revised accident analyses for the radiological consequences of a spent fuel cask drop accident and concludes that the licensee has adequately accounted for the effects of the proposed EPU on these analyses. The NRC staff further concludes that the plant site and the dose-mitigating ESFs remain acceptable with respect to the radiological consequences of a postulated spent fuel cask drop accident since the calculated whole-body and thyroid doses at the EAB and the LPZ outer boundary are well within the exposure guideline values of 10 CFR 100.11 and GDC-61. The NRC staff also concludes that the control room meets the dose requirements of GDC-19 for DBAs. Therefore, the NRC staff finds the licensee's proposed EPU acceptable with respect to spent fuel cask drop accidents.

[2.9.8 Additional Review Areas (Source Terms and Radiological Consequences Analyses)]

[Insert Regulatory Evaluation, Technical Evaluation, and Conclusion sections as necessary]

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FOR

SECTION 3.2 - BWR TEMPLATE SAFETY EVALUATION

2.10 Health Physics

2.10.1 Occupational and Public Radiation Doses

Regulatory Evaluation

The NRC staff conducts its review in this area to ascertain what overall effects the proposed EPU will have on both occupational and public radiation doses and to determine that the licensee has taken the necessary steps to ensure that any dose increases will be maintained as low as is reasonably achievable. The NRC staff's review includes an evaluation of any increases in radiation sources and how this may affect plant area dose rates, plant radiation zones, and plant area accessibility. The NRC staff evaluates how personnel doses needed to access plant vital areas following an accident are affected. The NRC staff considers the effects of the proposed EPU on nitrogen-16 levels in the plant and any effects this increase may have on radiation doses outside the plant and at the site boundary from skyshine. The NRC staff also considers the effects of the proposed EPU on plant effluent levels and any effect this increase may have on radiation doses at the site boundary. The NRC's acceptance criteria for occupational and public radiation doses are based on 10 CFR Part 20 and GDC-19. Specific review criteria are contained in SRP Sections 12.2, 12.3-12.4, and 12.5, and other guidance provided in Matrix 10 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on radiation source terms and plant radiation levels. The NRC staff concludes that the licensee has taken the necessary steps to ensure that any increases in radiation doses will be maintained as low as reasonably achievable. The NRC staff further concludes that the proposed EPU meets the requirements of 10 CFR Part 20 and GDC-19. Therefore, the NRC staff finds the licensee's proposed EPU acceptable with respect to radiation protection and ensuring that occupational radiation exposures will be maintained as low as reasonably achievable.

[2.10.2 Additional Review Areas (Health Physics)]

[Insert Regulatory Evaluation, Technical Evaluation, and Conclusion sections as necessary]

INSERT 11

FOR

SECTION 3.2 - BWR TEMPLATE SAFETY EVALUATION

2.11 Human Performance

2.11.1 Human Factors

Regulatory Evaluation

The area of human factors deals with programs, procedures, training, and plant design features related to operator performance during normal and accident conditions. The NRC staff's human factors evaluation is conducted to ensure that operator performance is not adversely affected as a result of system changes required for the proposed EPU. The NRC staff's review covers changes to operator actions, human-system interfaces, and procedures and training required for the proposed EPU. The NRC's acceptance criteria for human factors are based on GDC-19, 10 CFR 50.54(i) and (m), 10 CFR 50.59, 10 CFR 50.120, 10 CFR Part 55, and GL 82-33. Specific review criteria are contained in SRP Sections 13.2.1, 13.2.2, 13.5.2.1, and 18.0.

Technical Evaluation

The NRC staff has developed a standard set of questions for the review of the human factors area. The licensee has addressed these questions in its application. Following are the NRC staff's questions, the licensee's responses, and the NRC staff's determination of acceptability.

1. Changes in Emergency and Abnormal Operating Procedures

   Describe how the proposed EPU will change the plant emergency and abnormal operating procedures. (SRP Section 13.5.2.1)

[Insert licensee's response followed by additional NRC staff discussion if necessary]

2. Changes to Operator Actions Sensitive to Power Uprate

   Describe any new operator actions required as a result of the proposed EPU. Describe changes to any current operator actions related to emergency or abnormal operating procedures that will occur as a result of the proposed EPU. (SRP Section 18.0)

   (i.e., Identify and describe operator actions that will require additional response time or will have reduced time available. Your response should address any operator workarounds that might affect these response times. Identify any operator actions that are being automated or being changed from automatic to manual as a result of the power uprate. Provide justification for the acceptability of these changes).

[Insert licensee's response followed by additional NRC staff discussion if necessary]

3. Changes to Control Room Controls, Displays and Alarms

   Describe any changes the proposed EPU will have on the operator interfaces for control room controls, displays, and alarms. For example, what zone markings (e.g. normal, marginal and out-of-tolerance ranges) on meters will change? What setpoints will change?

How will the operators know of the change? Describe any controls, displays, alarms that will be upgraded from analog to digital instruments as a result of the proposed EPU and how operators were tested to determine they could use the instruments reliably. (SRP Section 18.0)

[Insert licensee's response followed by additional NRC staff discussion if necessary]

4. Changes on the Safety Parameter Display System

   Describe any changes the proposed EPU will have on the safety parameter display system. How will the operators know of the changes? (SRP Section 18.0)

[Insert licensee's response followed by additional NRC staff discussion if necessary]

5. Changes to the Operator Training Program and the Control Room Simulator

   Describe any changes the proposed EPU will have on the operator training program and the plant referenced control room simulator, and provide the implementation schedule for making the changes. (SRP Sections 13.2.1 and 13.2.2)

[Insert licensee's response followed by additional NRC staff discussion if necessary]

Conclusion

The NRC staff has reviewed the changes to operator actions, human-system interfaces, procedures, and training required for the proposed EPU and concludes that the licensee has (1) appropriately accounted for the effects of the proposed EPU on the available time for operator actions and (2) taken appropriate actions to ensure that operator performance is not adversely affected by the proposed EPU. The NRC staff further concludes that the licensee will continue to meet the requirements of GDC-19, 10 CFR 50.54(i) and (m), 10 CFR 50.59, 10 CFR 50.120, and 10 CFR Part 55 following implementation of the proposed EPU. Therefore, the NRC staff finds the licensee's proposed EPU acceptable with respect to the human factors aspects of the required system changes.

[2.11.2 Additional Review Areas (Human Performance)]

[Insert Regulatory Evaluation, Technical Evaluation, and Conclusion sections as necessary]

INSERT 12

FOR

SECTION 3.2 - BWR TEMPLATE SAFETY EVALUATION

2.12 Power Ascension and Testing Plan

2.12.1 Approach to EPU Power Level and Test Plan

Regulatory Evaluation

The purpose of the EPU test program is to demonstrate that SSCs will perform satisfactorily in service at the proposed EPU power level. The test program also provides additional assurance that the plant will continue to operate in accordance with design criteria at EPU conditions. The NRC staff's review includes an evaluation of: (1) plans for the initial approach to the proposed maximum licensed thermal power level, including verification of adequate plant performance, (2) transient testing necessary to demonstrate that plant equipment will perform satisfactorily at the proposed increased maximum licensed thermal power level, and (3) the test program's conformance with applicable regulations. The NRC's acceptance criteria for the EPU test program are based on 10 CFR Part 50, Appendix B, Criterion XI, as it relates to establishment of a test program to demonstrate that SSCs will perform satisfactorily in service. Specific review criteria are contained in SRP Section 14.2.1.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The staff has reviewed the EPU test program, including plans for the initial approach to the proposed maximum licensed thermal power level, transient testing necessary to demonstrate that plant equipment will perform satisfactorily at the proposed increased maximum licensed thermal power level, and the test program's conformance with applicable regulations. The staff concludes that the proposed EPU test program provides adequate assurance that the plant will operate in accordance with design criteria and that SSCs affected by the EPU or modified to support the proposed power increase will perform satisfactorily in service. Further, the staff finds that there is reasonable assurance that the EPU testing program satisfies the requirements of 10 CFR Part 50, Appendix B, Criterion XI. Therefore, the NRC staff finds the proposed EPU test program acceptable.

[2.12.2 Additional Review Areas (Power Ascension and Testing Plan)]

[Insert Regulatory Evaluation, Technical Evaluation, and Conclusion sections as necessary]

INSERT 13

FOR

SECTION 3.2 - BWR TEMPLATE SAFETY EVALUATION

2.13 Risk Evaluation

2.13.1 Risk Evaluation of Extended Power Uprate

Regulatory Evaluation

A risk evaluation is conducted to (1) demonstrate that the risks associated with the proposed EPU are acceptable and (2) determine if "special circumstances" are created by the proposed EPU. As described in Appendix D of SRP Chapter 19, special circumstances are any issues that would potentially rebut the presumption of adequate protection provided by the licensee to meet the deterministic requirements and regulations. The NRC staff's review covers the impact of the proposed EPU on core damage frequency (CDF) and large early release frequency (LERF) for the plant due to changes in the risks associated with internal events, external events, and shutdown operations. In addition, the NRC staff's review covers the quality of the risk analyses used by the licensee to support the application for the proposed EPU. This includes a review of licensee actions to address issues or weaknesses that may have been raised in previous NRC staff reviews of the licensee's individual plant examinations (IPEs) and individual plant examinations of external events (IPEEE), or by an industry peer review. The NRC's risk acceptability guidelines are contained in RG 1.174. Specific review guidance is contained in Matrix 13 of RS-001 and its attachments.

Technical Evaluation

[Insert technical evaluation]

Conclusion

The NRC staff has reviewed the licensee's assessment of the risk implications associated with the implementation of the proposed EPU and concludes that the licensee has adequately modeled and/or addressed the potential impacts associated with the implementation of the proposed EPU. The NRC staff further concludes that the results of the licensee's risk analysis indicate that the risks associated with the proposed EPU are acceptable and do not create the "special circumstances" described in Appendix D of SRP Chapter 19. Therefore, the NRC staff finds the risk implications of the proposed EPU acceptable.

[2.13.2 Additional Review Areas (Risk Evaluation)]

[Insert Regulatory Evaluation, Technical Evaluation, and Conclusion sections as necessary]

SECTION 3.3 of RS-001

TEMPLATE SAFETY EVALUATION

for

PRESSURIZED-WATER REACTOR EXTENDED POWER UPRATE

TABLE OF CONTENTS

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. ____ TO FACILITY OPERATING LICENSE NO. **[XXX-XX]**

[NAME OF LICENSEE]

[NAME OF FACILITY]

DOCKET NO. 50-**[XXX]**

1.0 INTRODUCTION

1.1 Application

By application dated [   ], as supplemented by letter**[s]** dated [   ], the **[Name of Licensee]** (the licensee) requested changes to the Facility Operating License and Technical Specifications (TSs) for the **[Plant Name]**. The supplemental letter[s] dated [   ], provided additional clarifying information that did not expand the scope of the initial application and did not change the Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on **[date] (XX FR XXXX).**

The proposed changes would increase the maximum steady-state reactor core power level from **[current licensed power level]** megawatts thermal (MWt) to **[power level proposed by the licensee]** MWt, which is an increase of approximately **[##]** percent. The proposed increase in power level is considered an extended power uprate (EPU).

1.2 Background

[Plant Name] is a pressurized-water reactor (PWR) plant of the **[Babcock & Wilcox (B&W), Combustion Engineering (CE), or Westinghouse 2-Loop, 3-Loop, or 4-Loop]** design with a **[######]** containment. **[Plant Name]** has the following special features/unique designs:

[Insert any special features/unique designs]

The NRC originally licensed **[Plant Name]** on **[date]** for operation at **[original licensed power level]** MWt. **[By Amendment No. [###] dated [   ], the NRC granted a power uprate to [Plant Name] of [##] percent, allowing the plant to be operated at [current licensed power level] MWt.] Therefore, the proposed EPU would result in an increase of approximately [##] percent over the original licensed power level [and [##] percent over the current licensed power level] for [Plant Name].]**

1.3 Licensee's Approach

The licensee's application for the proposed EPU follows the guidance in the Office of Nuclear Reactor Regulation's (NRR's) Review Standard (RS)-001, "Review Standard for Extended Power Uprates," to the extent that the review standard is consistent with the licensing basis of the plant. Where differences exist between the plant-specific licensing basis and RS-001, the licensee described the differences and provided evaluations consistent with the licensing basis of the plant. The licensee also used **[Identify topical reports or other documents used by the licensee for guidance related to the scope of the proposed EPU; NRC staff approvals, ranges of applicability, any limitations/restrictions associated with the documents; and consistency of the licensee's application with the ranges of applicability and limitations/restrictions. The discussion in this section is to cover topical reports and other documents referenced for the overall power uprate process. It is not intended to cover topical reports and other documents for specific methods of analyses. Topical reports and other documents referenced for specific methods of analyses are to be covered in the applicable technical evaluation section of this safety evaluation]**.

Insert this sentence if the licensee is planning to implement the EPU in one stage.
[The licensee plans to implement the EPU in one step. The licensee plans to make the modifications necessary to implement the EPU during the refueling outage in [season year (e.g., fall 2003)]. Subsequently, the plant will be operated at [##] MWt starting in Cycle [##].]

Insert this paragraph if the licensee is planning to implement the EPU in stages:
[The licensee plans to implement the EPU in [#] steps of [## and ##] percent. The licensee plans to make modifications necessary to implement the first step during the refueling outage in [season year (e.g., fall 2003)]. Subsequently, the plant will be operated at [##] MWt during Cycle [##]. The remainder of the modifications will be completed during the refueling outage in [season year (e.g., fall 2003)], with subsequent operation at [##] MWt starting in Cycle [##].]

1.4 Plant Modifications

The licensee has determined that several plant modifications are required to achieved the proposed EPU. The following is a list of these modifications and the licensee's proposed schedule for completing them.

[Provide a list of plant modifications.]

The NRC staff's evaluation of the licensee's proposed plant modifications is provided in Section 2.0 of this safety evaluation.

1.5 Method of NRC Staff Review

The NRC staff reviewed the licensee's application to ensure that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) activities proposed will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public. The purpose of the NRC staff's review is to evaluate the licensee's assessment of the impact of the proposed EPU on licensing-basis analyses. The NRC staff evaluated the licensee's application and supplements. The NRC staff also evaluated **[Include additional review items, as necessary (e.g., audits of certain information at the plant and vendor sites, and independent analyses), for areas where such analyses were deemed appropriate by the NRC staff]**.

In areas where the licensee and its contractors used NRC-approved or widely accepted methods in performing analyses related to the proposed EPU, the NRC staff reviewed relevant material to ensure that the licensee/contractor used the methods consistent with the limitations and restrictions placed on the methods. In addition, the NRC staff considered changes in plant operating conditions on the use of these methods to ensure that the methods are appropriate for use at the proposed EPU conditions. Details of the NRC staff's review are provided in Section 2.0 of this safety evaluation.

Audits of analyses supporting the EPU were conducted in relation to the following topics:

[Provide a list of areas for which audits were performed.]

The results of the audits are discussed in section 2.0 of this safety evaluation.

Independent NRC staff calculations were performed in relation to the following topics:

[Provide a list of areas for which independent NRC staff calculations were performed.]

The results of the calculations are discussed in section 2.0 of this safety evaluation.

2.0 EVALUATION

2.1 Materials and Chemical Engineering

INSERT 1 FOR SECTION 3.3 OF RS-001

2.2 Mechanical and Civil Engineering

INSERT 2 FOR SECTION 3.3 OF RS-001

2.3 Electrical Engineering

INSERT 3 FOR SECTION 3.3 OF RS-001

2.4 Instrumentation and Controls

INSERT 4 FOR SECTION 3.3 OF RS-001

2.5 Plant Systems

SEE INSERT 5 FOR SECTION 3.3 OF RS-001

2.6 Containment Review Considerations

SEE INSERT 6 FOR SECTION 3.3 OF RS-001

2.7 Habitability, Filtration, and Ventilation

SEE INSERT 7 FOR SECTION 3.3 OF RS-001

2.8 Reactor Systems

SEE INSERT 8 FOR SECTION 3.3 OF RS-001

2.9 Source Terms and Radiological Consequences Analyses

SEE INSERT 9 FOR SECTION 3.3 OF RS-001

2.10 Health Physics

SEE INSERT 10 FOR SECTION 3.3 OF RS-001

2.11 Human Performance

SEE INSERT 11 FOR SECTION 3.3 OF RS-001

2.12 Power Ascension and Testing Plan

SEE INSERT 12 FOR SECTION 3.3 OF RS-001

2.13 Risk Evaluation

SEE INSERT 13 FOR SECTION 3.3 OF RS-001

3.0 FACILITY OPERATING LICENSE AND TECHNICAL SPECIFICATION CHANGES

To achieve the EPU, the licensee proposed the following changes to the Facility Operating License and TSs for **[Plant Name]**.

[Provide a list of license and TSs changes (including license conditions) and an NRC staff evaluation of each.]

4.0 REGULATORY COMMITMENTS

Insert the following sentence if the licensee has not made any regulatory commitments in support of the EPU.
The licensee has made no regulatory commitments in its application for the EPU.

Insert the following if the licensee has made regulatory commitments in support of the EPU.
The licensee has made the following regulatory commitment(s):

[Provide a summary of each regulatory commitment made by the licensee.]

The NRC staff finds that reasonable controls for the implementation and for subsequent evaluation of proposed changes pertaining to the above regulatory commitment(s) are best provided by the licensee's administrative processes, including its commitment management program. The above regulatory commitments do not warrant the creation of regulatory requirements (items requiring prior NRC approval of subsequent changes).

5.0 RECOMMENDED AREAS FOR INSPECTION

The NRC staff has conducted an extensive review of the licensee's plans and analyses related to the proposed EPU and concluded that they are acceptable. The NRC review staff has identified the following areas for consideration by the NRC inspection staff during the licensee's implementation of the proposed EPU. These areas are recommended based on past experience with EPUs, the extent and unique nature of modifications required to achieve the proposed EPU, and new conditions of operation required for the proposed EPU. They do not constitute inspection requirements, but are intended to give inspectors insight into important bases for approving the EPU.

[Provide list of recommended areas for inspection.]

6.0 STATE CONSULTATION

In accordance with the Commission's regulations, the **[Name of State]** State official was notified of the proposed issuance of the amendment. The State official had **[no]** comments.
[If comments were received, address them here]

7.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, 51.33, and 51.35, a draft Environmental Assessment and finding of no significant impact was prepared and published in the *Federal Register* on **[Date (## FR #####)]**. The draft Environmental Assessment provided a 30-day opportunity for public comment. **[No]** comments were received on the draft Environmental Assessment.
[If comments were received, address them here.] The final Environmental Assessment was published in the *Federal Register* on **[Date (## FR #####)]**. Accordingly, based upon the environmental assessment, the Commission has determined that the issuance of this amendment will not have a significant effect on the quality of the human environment.

8.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

9.0 REFERENCES

1. RS-001, "Review Standard for Extended Power Uprates," Date.

2. **[Insert additional references as necessary]**

Attachment: List of Acronyms

Principal Contributors:

Date:

LIST OF ACRONYMS

AAC	alternate ac sources
ac	alternating current
ALARA	as low as reasonably achievable
ARAVS	auxiliary and radwaste area ventilation system
ARI	alternate rod insertion
ASME	American Society of Mechanical Engineers
ATWS	anticipated transient without scram
B&PV	boiler and pressure vessel
B&W	Babcock and Wilcox
BL	bulletin
BOP	balance-of-plant
BRS	boron recovery system
BTP	branch technical position
CDF	core damage frequency
CE	Combustion Engineering
CFR	*Code of Federal Reguations*
CFS	condensate and feedwater system
CRAVS	control room area ventilation system
CRDM	control rod drive mechanism
CRDS	control rod drive system
CUF	cumulative usage factor
CVCS	chemical and volume control system
CWS	circulating water system
DBA	design-basis accident
DBLOCA	design-basis loss-of-coolant accident
dc	direct current
DG	draft guide
DSS	diverse scram system

EAB	exclusion area boundary
ECCS	emergency core cooling system
EFDS	equipment and floor drainage system
EPG	emergency procedure guideline
EPRI	Electric Power Research Institute
EPU	extended power uprate
EQ	environmental qualification
ESF	engineered safety feature
ESFAS	engineered safety feature actuation system
ESFVS	engineered safety feature ventilation system
FAC	flow-accelerated corrosion
FHA	fuel handling accident
FPP	fire protection program
GDC	general design criterion
GL	generic letter
I&C	instrumentation and controls
IN	information notice
IPE	individual plant examination
IPEEE	individual plant examination of external events
LERF	large early release frequency
LLHS	light load handling system
LOCA	loss-of-coolant accident
LOOP	loss of offsite power
LPZ	low population zone
MC	main condenser
MCES	main condenser evacuation system
MOV	motor-operated valve
MSLB	main steamline break

MSSS	main steam supply system
MTC	moderator temperature coefficient
MWt	megawatts thermal
NEI	Nuclear Energy Institute
NPSH	net positive suction head
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
NSSS	nuclear steam supply system
O&M	operations and maintenance
P-T	pressure-temperature
PRT	pressurizer relief tank
PWR	pressurized-water reactor
PWSCC	primary water stress-corrosion cracking
RCPB	reactor coolant pressure boundary
RCS	reactor coolant system
REA	rod ejection accident
RG	regulatory guide
RHR	residual heat removal
RS	review standard
SAFDL	specified acceptable fuel design limit
SAG	severe accident guideline
SAR	Safety Analysis Report
SBO	station blackout
SFP	spent fuel pool
SFPAVS	spent fuel pool area ventilation system
SG	steam generator
SGBS	steam generator blowdown system
SGTR	steam generator tube rupture

SRP	Standard Review Plan
SSCs	structures, systems, and components
SSE	safe-shutdown earthquake
SWMS	solid waste management system
SWS	service water system
TAVS	turbine area ventilation system
TBS	turbine bypass system
TCV	turbine control valve
TEDE	total effective dose equivalent
TS	technical specification
UHS	ultimate heat sink

INSERT 1

FOR

SECTION 3.3 - PWR TEMPLATE SAFETY EVALUATION

2.1 Materials and Chemical Engineering

2.1.1 Reactor Vessel Material Surveillance Program

Regulatory Evaluation

The reactor vessel material surveillance program provides a means for determining and monitoring the fracture toughness of the reactor vessel beltline materials to support analyses for ensuring the structural integrity of the ferritic components of the reactor vessel. The NRC staff's review primarily focuses on the effects of the proposed EPU on the licensee's reactor vessel surveillance capsule withdrawal schedule. The NRC's acceptance criteria are based on (1) General Design Criterion (GDC)-14 for assuring an extremely low probability of rapidly propagating fractures of the reactor coolant pressure boundary (RCPB); (2) GDC-31 for assuring that the RCPB will behave in a nonbrittle manner and the probability of a rapidly propagating fracture is minimized; (3) 10 CFR Part 50, Appendix H, for determination and monitoring of fracture toughness; and (4) 10 CFR 50.60 for compliance with the requirements of 10 CFR Part 50, Appendix H. Specific review criteria are contained in Standard Review Plan (SRP) Section 5.3.1 and other guidance provided in Matrix 1 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed EPU on the reactor vessel surveillance withdrawal schedule and concludes that the licensee has adequately addressed changes in neutron fluence and their effects on the schedule. The NRC staff further concludes that the reactor vessel capsule withdrawal schedule is appropriate to ensure that the material surveillance program will continue to meet the requirements of GDC-14, GDC-31, 10 CFR Part 50, Appendix H, and 10 CFR 50.60 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the reactor vessel material surveillance program.

2.1.2 Pressure-Temperature Limits and Upper-Shelf Energy

Regulatory Evaluation

Pressure-temperature (P-T) limits are established to ensure the structural integrity of the ferritic components of the RCPB during any condition of normal operation, including anticipated operational occurrences and hydrostatic tests. The NRC staff's review of P-T limits covers the P-T limits' methodology and the calculations for the specified effective full power years, considering neutron embrittlement effects and using linear elastic fracture mechanics. The NRC's acceptance criteria for P-T limits are based on (1) GDC-14 for assuring an extremely low probability of abnormal leakage, rapidly propagating failure, and gross rupture of the RCPB; (2) GDC-31 for assuring that the RCPB will behave in a nonbrittle manner and the probability of a rapidly propagating fracture is minimized; (3) 10 CFR Part 50, Appendix G, for material testing and fracture toughness; and (4) 10 CFR 50.60 for compliance with the requirements of 10 CFR Part 50, Appendix G. Specific review criteria are contained in SRP Section 5.3.2 and other guidance provided in Matrix 1 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed EPU on the P-T limits for the plant and concludes that the licensee has adequately addressed changes in neutron fluence and their effects on the P-T limits. The NRC staff further concludes that the licensee has demonstrated the validity of the proposed P-T limits for the proposed EPU operation. Based on this, the NRC staff concludes that the proposed P-T limits will continue to meet the requirements of GDC-14, GDC-31, 10 CFR Part 50, Appendix G, and 10 CFR 50.60 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the proposed P-T limits.

2.1.3 Pressurized Thermal Shock

Regulatory Evaluation

The pressurized thermal shock (PTS) evaluation provides a means for assessing the susceptibility of the reactor vessel beltline materials to PTS events to assure that adequate fracture toughness is provided for supporting reactor operation. The NRC staff's review covers the PTS methodology and the calculations for the reference temperature, RT_{PTS}, at the expiration of the license, considering neutron embrittlement effects. The NRC's acceptance criteria for PTS are based on (1) GDC-14 for assuring an extremely low probability of an abnormal leakage, a rapidly propagating failure, or a gross rupture of the RCPB; (2) GDC-31 for assuring that the RCPB will behave in a nonbrittle manner, and the probability of a rapidly propagating fracture is minimized; and (3) 10 CFR 50.61 for fracture toughness criteria for PTS events. Specific review criteria are contained in SRP Section 5.3.2 and other guidance provided in Matrix 1 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed EPU on the PTS for the plant and concludes that the licensee has adequately addressed changes in neutron fluence and their effects on PTS. The NRC staff further concludes that the licensee has demonstrated that the plant will continue to meet the requirements of GDC-14, GDC-31, and 10 CFR 50.61 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to PTS.

2.1.4 Reactor Internal and Core Support Materials

Regulatory Evaluation

The reactor internals and core supports include structures, systems, and components (SSCs) that perform safety functions and/or whose failure could affect safety functions performed by other SSCs. These safety functions include reactivity monitoring and control, core cooling, and fission product confinement (within both the fuel cladding and the reactor coolant system (RCS)). The NRC staff's review covers the materials' specifications and mechanical properties, welds, weld controls, nondestructive examination procedures, corrosion resistance, and susceptibility to degradation. The NRC's acceptance criteria for reactor internal and core support materials are based on GDC-1 and 10 CFR 50.55a for material specifications, controls on welding, and inspection of reactor internals and core supports. Specific review criteria are contained in SRP Section 4.5.2, WCAP-14277, and BAW-2248.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed EPU on the susceptibility of reactor internal and core support materials to known degradation mechanisms and concludes that the licensee has identified appropriate degradation management programs to address the effects of changes in operating temperature and neutron fluence on the integrity of reactor internal and core support materials. The NRC staff further concludes that the licensee has demonstrated that the reactor internal and core support materials will continue to be acceptable and will continue to meet the requirements of GDC-1 and 10 CFR 50.55a following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to reactor internal and core support materials.

2.1.5 Reactor Coolant Pressure Boundary Materials

Regulatory Evaluation

The RCPB defines the boundary of systems and components containing the high pressure fluids produced in the reactor. The NRC staff's review of RCPB materials covers their specifications, compatibility with the reactor coolant, fabrication and processing, susceptibility to degradation, and degradation management programs. The NRC's acceptance criteria for RCPB materials are based on (1) GDC-1 and 10 CFR 50.55a for quality standards; (2) GDC-4 for compatibility of components with environmental conditions; (3) GDC-14 and GDC-31 for assuring an extremely low probability of a rapidly propagating fracture or a gross rupture of the RCPB; and (4) 10 CFR Part 50, Appendix G, for materials testing and acceptance criteria for fracture toughness of the RCPB. Specific review criteria are contained in SRP Section 5.2.3 and other guidance provided in Matrix 1 of RS-001. Additional review guidance for primary water stress corrosion cracking (PWSCC) of dissimilar metal welds and associated inspection programs is contained in Generic Letter (GL) 97-01, Information Notice (IN) 00-17, Bulletin (BL) 01-01, BL 02-01, and BL 02-02. Additional review guidance for thermal embrittlement of cast austenitic stainless steel components is contained in a letter from C. Grimes, NRC, to D. Walters, Nuclear Energy Institute (NEI), dated May 19, 2000.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed EPU on the susceptibility of RCPB materials to known degradation mechanisms and concludes that the licensee has identified appropriate degradation management programs to address the effects of changes in system operating temperature on the integrity of RCPB materials. The NRC staff further concludes that the licensee has demonstrated that the RCPB materials will continue to be acceptable following implementation of the proposed EPU and will continue to meet the requirements of GDC-1, GDC-4, GDC-14, GDC-31, 10 CFR Part 50, Appendix G, and 10 CFR 50.55a. Therefore, the NRC staff finds the proposed EPU acceptable with respect to RCPB materials.

2.1.6 Leak-Before-Break

Regulatory Evaluation

Leak-before-break (LBB) analyses provide a means for eliminating from the design basis the dynamic effects of the postulated pipe ruptures for a piping system. NRC approval of LBB for a plant permits the licensee to (1) remove protective hardware along the piping system (e.g., pipe whip restraints and jet impingement barriers) and (2) redesign pipe-connected components, their supports and their internals. The NRC staff's review for LBB covers (a) direct pipe failure mechanisms (e.g., water hammer, creep damage, erosion, corrosion, fatigue, and environmental conditions); (b) indirect pipe failure mechanisms (e.g., seismic events, system overpressurizations, fires, flooding, missiles, and failures of SSCs in close proximity to the piping); and (c) the deterministic fracture mechanics and leak detection methods. The NRC's acceptance criteria for LBB are based on GDC-4 for exclusion of dynamic effects of the postulated pipe ruptures. Specific review criteria are contained in draft SRP Section 3.6.3 and other guidance provided in Matrix 1 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed EPU on the LBB analysis for the plant and concludes that the licensee has adequately addressed changes in primary system pressure and temperature and their effects on the LBB analyses. The NRC staff further concludes that the licensee has demonstrated that the LBB analyses will continue to be valid following implementation of the proposed EPU and that lines for which the licensee credits LBB will continue to meet the requirements of GDC-4. Therefore, the NRC staff finds the proposed EPU acceptable with respect to LBB.

2.1.7 Protective Coating Systems (Paints) - Organic Materials

Regulatory Evaluation

Protective coating systems (paints) provide a means for protecting the surfaces of facilities and equipment from corrosion and contamination from radionuclides and also provide wear protection during plant operation and maintenance activities. The NRC staff's review covers protective coating systems used inside the containment for their suitability for and stability under design-basis accident (DBA) conditions considering radiation and chemical effects. The NRC's acceptance criteria for protective coating systems are based on (1) 10 CFR Part 50, Appendix B, for the quality assurance requirements for the design, fabrication, and construction of safety-related SSCs and (2) Regulatory Guide 1.54, Revision 1, for application and performance monitoring of coatings in nuclear power plants. Specific review criteria are contained in SRP Section 6.1.2.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed EPU on protective coating systems and concludes that the licensee has appropriately addressed changes in conditions following a design-basis loss-of-coolant accident (LOCA) and their effects on the protective coatings. The NRC staff further concludes that the licensee has demonstrated that the protective coatings will continue to be acceptable following implementation of the proposed EPU and will continue to meet the requirements of 10 CFR Part 50, Appendix B. Therefore, the NRC staff finds the proposed EPU acceptable with respect to protective coatings systems.

2.1.8 Flow-Accelerated Corrosion

Regulatory Evaluation

Flow-accelerated corrosion (FAC) is a corrosion mechanism occurring in carbon steel components exposed to flowing single- or two-phase water. The components made from stainless steel are immune to FAC, and FAC is significantly reduced in components containing small amounts of chromium or molybdenum. The rates of material loss by FAC depend on velocity of flow, temperature, steam quality, oxygen content, and pH. During plant operation, control of these parameters is limited and the optimum conditions for minimizing FAC effects, in most cases, cannot be achieved. Loss of material by FAC will, therefore, occur. The NRC staff reviews the effects of the proposed EPU on FAC and the adequacy of the licensee's FAC program to predict the rate of loss so that repair or replacement of damaged components could be made before they reach critical thickness. The licensee's FAC program is based on NUREG-1344, GL 89-08, and the guidelines in the Electric Power Research Institute (EPRI) report NSAC-202L-R2. It consists of predicting loss of material using the CHECWORKS computer code, and visual inspection and volumetric examination of the affected components. The NRC's acceptance criteria are based on the structural evaluation of the minimum acceptable wall thickness for the components undergoing degradation by FAC.

Technical Evaluation

[Insert technical evaluation.]

Conclusions

The NRC staff has reviewed the licensee's evaluation of the effect of the proposed EPU on the FAC analysis for the plant and concludes that the licensee has adequately addressed the impact of changes in the plant operating conditions on the FAC analysis. Further, the NRC staff concludes that the licensee has demonstrated that the updated analyses will predict the loss of material by FAC and will ensure timely repair or replacement of degraded components following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to FAC.

2.1.9 Steam Generator Tube Inservice Inspection

Regulatory Evaluation

Steam generator (SG) tubes constitute a large part of the RCPB. SG tube inservice inspection (ISI) provides a means for assessing the structural and leaktight integrity of the SG tubes through periodic inspection and testing of critical areas and features of the tubes. The NRC staff's review in this area covers the effects of changes in differential pressure, temperature, and flow rates from the proposed EPU on plugging limits, potential degradation mechanisms (e.g., flow induced vibration), and plant-specific alternate repair criteria and redefined inspection boundaries. The NRC's acceptance criteria for SG tube ISI are based on 10 CFR 50.55a for periodic inspection and testing of the RCPB. Specific review criteria are contained in SRP Section 5.4.2.2 and other guidance provided in Matrix 1 of RS-001. Additional review guidance is contained in **[provide specific plant technical specification]** for SG surveillance requirements, Regulatory Guide 1.121 for SG tube plugging limits, GL 95-03 and Bulletin 88-02 for degradation mechanisms, NEI 97-06 for structural and leakage performance criteria, and **[provide topical reports approved for the plant]** that form the basis for alternate repair criteria or redefined inspection boundaries.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed EPU on SG tube integrity and concludes that the licensee has adequately assessed the acceptability of the plant's TSs and has identified appropriate degradation management inspections to address the effects of changes in temperature, differential pressure, and flow rates on the SG tube integrity. The NRC staff further concludes that licensee has demonstrated that SG tube integrity will continue to be maintained and will continue to meet the requirements of 10 CFR 50.55a and the performance criteria in NEI 97-06 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to SG tube ISI.

2.1.10 Steam Generator Blowdown System

Regulatory Evaluation

Control of secondary side water chemistry is important for preventing degradation of steam generator tubes. The steam generator blowdown system (SGBS) provides a means for removing steam generator secondary-side impurities and thus assists in maintaining acceptable secondary-side water chemistry in the steam generators. The design basis of the SGBS includes consideration of expected and design flows for all modes of operation. The NRC staff's review covers the ability of the SGBS to remove particulate and dissolved impurities from the steam generator secondary side during normal operation, including anticipated operational occurrences (main condenser inleakage and primary-to-secondary leakage). The NRC's acceptance criteria for the SGBS are based on GDC-14 for secondary water chemistry control to ensure the integrity of RCPB material. Specific review criteria are contained in SRP Section 10.4.8.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed EPU on the SGBS and concludes that the licensee has adequately addressed changes in system flow and impurity levels and their effects on the SGBS. The NRC staff further concludes that the licensee has demonstrated that the SGBS will continue to be acceptable and will continue to meet the requirements of GDC-14 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to SGBS.

2.1.11 Chemical and Volume Control System

Regulatory Evaluation

The chemical and volume control system (CVCS) and boron recovery system (BRS) provide means for (a) maintaining the required water inventory and quality in the RCS, (b) supplying seal-water flow to the reactor coolant pumps and pressurizer auxiliary spray, (c) controlling the boron neutron absorber concentration in the reactor coolant, (d) controlling the primary water chemistry and reducing coolant radioactivity level, and (e) supplying recycled coolant for demineralized water makeup for normal operation and high pressure injection flow to the emergency core cooling system (ECCS) in the event of postulated accidents. The NRC staff reviewed the safety-related functional performance characteristics of CVCS components. The NRC's acceptance criteria are based on (1) GDC-14 for assuring RCPB material integrity by means of the CVCS being capable of maintaining RCS water chemistry necessary to meet RCS water chemistry TSs, and (2) GDC-29 for the reliability of the CVCS to provide negative reactivity to the reactor by supplying borated water to the RCS in the event of anticipated operational occurrences. Specific review criteria are contained in SRP Section 9.3.4.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed EPU on the CVCS and BRS and concludes that the licensee has adequately addressed changes in the temperature of the reactor coolant and their effects on the CVCS and BRS. The NRC staff further concludes that the licensee has demonstrated that the CVCS and BRS will continue to be acceptable and will continue to meet the requirements of GDC-14 and GDC-29 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the CVCS.

[2.1.12 <u>Additional Review Areas</u> (Materials and Chemical Engineering)]

[Insert Regulatory Evaluation, Technical Evaluation, and Conclusion sections as necessary]

INSERT 2

FOR

SECTION 3.3 - PWR TEMPLATE SAFETY EVALUATION

2.2 Mechanical and Civil Engineering

2.2.1 Pipe Rupture Locations and Associated Dynamic Effects

Regulatory Evaluation

SSCs important to safety could be impacted by the pipe-whip dynamic effects of a pipe rupture. The NRC staff conducts a review of pipe rupture analyses to ensure that SSCs important to safety are adequately protected from the effects of pipe ruptures. The NRC staff's review covers (1) the implementation of criteria for defining pipe break and crack locations and configurations, (2) the implementation of criteria dealing with special features, such as augmented inservice inspection programs or the use of special protective devices such as pipe-whip restraints, (3) the pipe-whip dynamic analyses and results, including the jet thrust and impingement forcing functions and pipe-whip dynamic effects, and (4) the design adequacy of supports for SSCs provided to ensure that the intended design functions of the SSCs will not be impaired to an unacceptable level as a result of pipe-whip or jet impingement loadings. The NRC staff's review is focused on the effects that the proposed EPU may have on items (1) thru (4) above. The NRC's acceptance criteria are based on GDC-4 as related to SSCs important to safety being designed to accommodate the dynamic effects of a postulated pipe rupture. Specific review criteria are contained in SRP Section 3.6.2.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's evaluations related to determinations of rupture locations and associated dynamic effects and concludes that the licensee has adequately addressed the effects of the proposed EPU on them. The NRC staff further concludes that the licensee has demonstrated that SSCs important to safety will continue to meet the requirements of GDC-4 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the determination of rupture locations and dynamic effects associated with the postulated rupture of piping.

2.2.2 Pressure-Retaining Components and Component Supports

Regulatory Evaluation

The NRC staff's review concerns the structural integrity of pressure-retaining components (and their supports) designed in accordance with the American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (B&PV Code), Section III, Division 1, and GDCs 1, 2, 4, 14, and 15. The NRC staff's review is focused on the effects of the proposed EPU on the design input parameters and the design-basis loads and load combinations for normal operating, upset, emergency, and faulted conditions. The NRC staff's review covers (1) the analyses of flow-induced vibration and (2) the analytical methodologies, assumptions, ASME Code editions, and computer programs used for these analyses. The NRC staff's review also includes a comparison of the resulting stresses and cumulative fatigue usage factors (CUFs) against the code-allowable limits. The NRC's acceptance criteria are based on (1) 10 CFR 50.55a and GDC-1 as they relate to SSCs being designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed; (2) GDC-2 as it relates to SSCs important to safety being designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions; (3) GDC-4 as it relates to SSCs important to safety being designed to accommodate the effects of and to be compatible with the environmental conditions of normal and accident conditions; (4) GDC-14 as it relates to the RCPB being designed, fabricated, erected, and tested to have an extremely low probability of abnormal leakage, rapidly propagating failure, and gross rupture; and (5) GDC-15 as it relates to the RCS being designed with sufficient margin to ensure that the design conditions are not exceeded. Specific review criteria are contained in SRP Sections 3.9.1, 3.9.2, 3.9.3, and 5.2.1.1; and other guidance provided in Matrix 2 of RS-001.

Technical Evaluation

Nuclear Steam Supply System Piping, Components, and Supports

[Insert technical evaluation for Nuclear Steam Supply System piping, components, and supports.]

Balance-of-Plant Piping, Components, and Supports

[Insert technical evaluation for balance-of-plant piping, components, and supports.]

Reactor Vessel and Supports

[Insert technical evaluation for reactor vessel and supports.]

Control Rod Drive Mechanism

[Insert technical evaluation for control rod drive mechanism.]

Steam Generators and Supports

[Insert technical evaluation for steam generators and supports.]

Reactor Coolant Pumps and Supports

[Insert technical evaluation for reactor coolant pumps and supports.]

Pressurizer and Supports

[Insert technical evaluation for pressurizer and supports.]

Conclusion

The NRC staff has reviewed the licensee's evaluations related to the structural integrity of pressure-retaining components and their supports and concludes that the licensee has adequately addressed the effects of the proposed EPU on them. The NRC staff further concludes that the licensee has demonstrated that pressure-retaining components and their supports will continue to meet the requirements of 10 CFR 50.55a, GDC-1, GDC-2, GDC-4, GDC-14, and GDC-15 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the structural integrity of the pressure-retaining components and their supports.

2.2.3 Reactor Pressure Vessel Internals and Core Supports

Regulatory Evaluation

Reactor pressure vessel internals consist of all the structural and mechanical elements inside the reactor vessel, including core support structures. The NRC staff reviews the effects of the proposed EPU on the design input parameters and the design-basis loads and load combinations for the reactor internals for normal operation, upset, emergency, and faulted conditions. These include pressure differences and thermal effects for normal operation, transient pressure loads associated with LOCAs, and the identification of design transient occurrences. The NRC staff's review covers (1) the analyses of flow-induced vibration for safety-related and non-safety-related reactor internal components and (2) the analytical methodologies, assumptions, ASME Code editions, and computer programs used for these analyses. The NRC staff's review also includes a comparison of the resulting stresses and CUFs against the corresponding Code-allowable limits. The NRC's acceptance criteria are based on (1) GDC-1 and 10 CFR 50.55a for the design of reactor internals using quality standards commensurate with the importance of the safety functions to be performed; (2) GDC-2 for the design of reactor internals to withstand the effects of earthquakes without the loss of capability to perform their safety functions; (3) GDC-4 for the design of reactor internals to accommodate the effects of and to be compatible with the environmental conditions associated with normal operations, maintenance, testing, and postulated LOCAs; and (4) GDC-10 for the design of reactor internals with appropriate margin to assure that specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences. Specific review criteria are contained in SRP Sections 3.9.1, 3.9.2, 3.9.3, and 3.9.5; and other guidance provided in Matrix 2 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's evaluations related to the structural integrity of reactor internals and core supports and concludes that the licensee has adequately addressed the effects of the proposed EPU on them. The NRC staff further concludes that the licensee has demonstrated that the reactor internals and core supports will continue to meet the requirements of 10 CFR 50.55a, GDC-1, GDC-2, GDC-4, and GDC-10 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the design of the reactor internal and core supports.

2.2.4 Safety-Related Valves and Pumps

Regulatory Evaluation

The NRC's staff's review includes certain safety-related pumps and valves typically designated as Class 1, 2, or 3 under Section III of the ASME B&PV Code and within the scope of Section XI of the ASME B&PV Code and the ASME Operations and Maintenance (O&M) Code, as applicable. The NRC staff's review focuses on the effects of the proposed EPU on the required functional performance of the valves and pumps. The review also covers any impacts that the proposed EPU may have on the licensee's motor-operated valve (MOV) programs related to GL 89-10, GL 96-05, and GL 95-07. The NRC staff also evaluates the licensee's consideration of lessons learned from the MOV program and the application of those lessons learned to other safety-related power-operated valves. The NRC's acceptance criteria are based on (1) GDC-1 for testing components important to safety to quality standards commensurate with the importance of the safety functions to be performed; (2) GDC-37, GDC-40, GDC-43, and GDC-46 for periodic functional testing of the emergency core cooling system, the containment heat removal system, the containment atmospheric cleanup systems, and the cooling water system, respectively, to ensure the leak-tight integrity and performance of their active components; (3) GDC-54 for piping systems penetrating containment being designed with the capability to periodically test the operability of the isolation and determine valve leakage acceptability; and (4) 10 CFR 50.55a(f) for including pumps and valves whose function is required for safety in the inservice testing program to verify operational readiness by periodic testing. Specific review criteria are contained in SRP Sections 3.9.3 and 3.9.6; and other guidance provided in Matrix 2 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's evaluations related to the functional performance of safety-related valves and pumps and concludes that the licensee has adequately addressed the effects of the proposed EPU on them. The NRC staff further concludes that the licensee has adequately evaluated the effects of the proposed EPU on its MOV programs related to GL 89-10, GL 96-05, and GL 95-07, and the lessons learned from those programs to other safety-related power-operated valves. Based on this, the NRC staff concludes that the licensee has demonstrated that safety-related valves and pumps will continue to meet the requirements of GDC-1, GDC-37, GDC-40, GDC-43, GDC-46, GDC-54, and 10 CFR 50.55a(f) following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to safety-related valves and pumps.

2.2.5 Seismic and Dynamic Qualification of Mechanical and Electrical Equipment

Regulatory Evaluation

Mechanical and electrical equipment covered by this section includes equipment associated with systems that are essential to emergency reactor shutdown, containment isolation, reactor core cooling, and containment and reactor heat removal. Equipment associated with systems essential in preventing significant release of radioactive materials to the environment are also covered by this section. The NRC staff's review focuses on the effects of the proposed EPU on the qualification of the equipment to withstand seismic events and the dynamic effects associated pipe-whip and jet impingement forces. The primary input motions due to the safe shutdown earthquake (SSE) are not affected by an EPU. The NRC's acceptance criteria are based on (1) GDC-1 and GDC-30 for qualifying equipment to appropriate quality standards commensurate with the importance of the safety functions to be performed; (2) GDC-2 and Appendix A to 10 CFR Part 100 for qualifying equipment to withstand the effects of natural phenomena, such as earthquakes; (3) GDC-4 for qualifying equipment to withstand the dynamic effects associated with external missiles and internally generated missiles, pipe whip, and jet impingement forces; (4) GDC-14 for qualifying equipment associated with the RCPB to ensure an extremely low probability of abnormal leakage, rapidly propagating failure. and gross rupture; and (5) Appendix B to 10 CFR Part 50 for the quality assurance requirements for qualification of equipment. Specific review criteria are contained in SRP Section 3.10.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's evaluations of the effects of the proposed EPU on the qualification of mechanical and electrical equipment and concludes that the licensee has (1) adequately addressed the effects of the proposed EPU on this equipment and (2) demonstrated that the equipment will continue to meet the requirements of GDCs 1, 2, 4, 14, and 30; 10 CFR Part 100, Appendix A; and 10 CFR Part 50, Appendix B, following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the qualification of the mechanical and electrical equipment.

[2.2.6 Additional Review Areas (Mechanical and Civil Engineering)]

[Insert Regulatory Evaluation, Technical Evaluation, and Conclusion sections as necessary]

INSERT 3

FOR

SECTION 3.3 - PWR TEMPLATE SAFETY EVALUATION

2.3 Electrical Engineering

2.3.1 Environmental Qualification of Electrical Equipment

Regulatory Evaluation

Environmental qualification (EQ) of electrical equipment involves demonstrating that the equipment is capable of performing its safety function under significant environmental stresses which could result from DBAs. The NRC staff's review is focused on the effects of the proposed EPU on the environmental conditions that the electrical equipment will be exposed to during normal operation, anticipated operational occurrences, and accidents. The NRC staff's review is conducted to ensure that the electrical equipment will continue to be capable of performing its safety functions following implementation of the proposed EPU. The NRC's acceptance criteria for EQ of electrical equipment are based on 10 CFR 50.49 as it relates to the qualification of electrical equipment important to safety that is located in a harsh environment. Specific review criteria are contained in SRP Section 3.11.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the EQ of electrical equipment and concludes that the licensee has adequately addressed the effects of the proposed EPU on the environmental conditions for and the qualification of the electrical equipment. The NRC staff further concludes that the electrical equipment will continue to meets the relevant requirements of 10 CFR 50.49 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the EQ of electrical equipment.

2.3.2 Offsite Power System

Regulatory Evaluation

The offsite power system includes two or more physically independent circuits capable of operating independently of the onsite standby power sources. The NRC staff's review covers the descriptive information, analyses, and referenced documents for the offsite power system; and the stability studies for the electrical transmission grid. The NRC staff's review is focused on the requirement that loss of the nuclear unit, the largest operating unit on the grid, or the most critical transmission line will not result in the loss of offsite power to the plant following implementation of the proposed EPU. The NRC's acceptance criteria for offsite power systems are based on GDC-17. Specific review criteria are contained in SRP Sections 8.1 and 8.2, Appendix A to SRP Section 8.2, and Branch Technical Positions (BTPs) PSB-1 and ICSB-11.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the offsite power system and concludes that the offsite power system will continue to meet the requirements of GDC-17 following implementation of the proposed EPU. Adequate physical and electrical separation exists and the offsite power system has the capacity and capability to supply power to all safety loads and other required equipment. The NRC staff further concludes that the impact of the proposed EPU on grid stability is insignificant. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the offsite power system.

2.3.3 AC Onsite Power System

Regulatory Evaluation

The ac onsite power system includes those standby power sources, distribution systems, and auxiliary supporting systems provided to supply power to safety-related equipment. The NRC staff's review covers the descriptive information, analyses, and referenced documents for the ac onsite power system. The NRC's acceptance criteria for the ac onsite power system are based on GDC-17 for the capability of the ac onsite power system to perform its intended functions during all plant operating and accident conditions. Specific review criteria are contained in SRP Sections 8.1 and 8.3.1.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the ac onsite power system and concludes that the licensee has adequately accounted for the effects of the proposed EPU on the system's functional design. The NRC staff further concludes that the ac onsite power system will continue to meet the requirements of GDC-17 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the ac onsite power system.

2.3.4 DC Onsite Power System

Regulatory Evaluation

The dc onsite power system includes the dc power sources and their distribution and auxiliary supporting systems that are provided to supply motive or control power to safety-related equipment. The NRC staff's review covers the information, analyses, and referenced documents for the dc onsite power system. The NRC's acceptance criteria for the dc onsite power system are based on GDC-17 for the capability of the dc onsite power system to facilitate the functioning of SSCs important to safety. Specific review criteria are contained in SRP Sections 8.1 and 8.3.2

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the dc onsite power system and concludes that the licensee has adequately accounted for the effects of the proposed EPU on the system's functional design. The NRC staff further concludes that the dc onsite power system will continue to meet the requirements of GDC-17 following implementation of the proposed EPU. Adequate physical and electrical separation exists and the system has the capacity and capability to supply power to all safety loads and other required equipment. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the dc onsite power system.

2.3.5 Station Blackout

Regulatory Evaluation

Station blackout (SBO) refers to a complete loss of ac electric power to the essential and nonessential switchgear buses in a nuclear power plant. SBO involves the loss of offsite power concurrent with a turbine trip and failure of the onsite emergency ac power system. SBO does not include the loss of available ac power to buses fed by station batteries through inverters or the loss of power from "alternate ac sources" (AACs). The NRC staff's review focuses on the impact of the proposed EPU on the plant's ability to cope with and recover from a SBO event for the period of time established in the plant's licensing basis. The NRC's acceptance criteria for SBO are based on 10 CFR 50.63. Specific review criteria are contained in SRP Sections 8.1 and Appendix B to SRP Section 8.2; and other guidance provided in Matrix 3 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the plant's ability to cope with and recover from a SBO event for the period of time established in the plant's licensing basis. The NRC staff concludes that the licensee has adequately evaluated the effects of the proposed EPU on SBO and demonstrated that the plant will continue to meet the requirements of 10 CFR 50.63 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to SBO.

[2.3.6 Additional Review Areas (Electrical Engineering)]

[Insert Regulatory Evaluation, Technical Evaluation, and Conclusion sections as necessary]

INSERT 4

FOR

SECTION 3.3 - PWR TEMPLATE SAFETY EVALUATION

2.4 Instrumentation and Controls

2.4.1 Reactor Protection, Safety Features Actuation, and Control Systems

Regulatory Evaluation

Instrumentation and control systems are provided (1) to control plant processes having a significant impact on plant safety, (2) to initiate the reactivity control system (control rods), (3) to initiate the engineered safety features (ESF) systems and essential auxiliary supporting systems, and (4) for use to achieve and maintain a safe shutdown condition of the plant. Diverse instrumentation and control systems and equipment are provided for the express purpose of protecting against potential common-mode failures of instrumentation and control protection systems. The NRC staff conducts a review of the reactor trip system, engineered safety feature actuation system (ESFAS), safe shutdown systems, control systems, and diverse instrumentation and control systems for the proposed EPU to ensure that the systems and any changes required for the proposed EPU are adequately designed such that the systems continue to meet their safety functions. The NRC staff's review is also conducted to ensure that failures of the systems do not affect safety functions. The NRC's acceptance criteria related to the quality of design of protection and control systems are based on 10 CFR 50.55a(a)(1), 10 CFR 50.55a(h), and GDCs 1, 4, 13, 19, 20, 21, 22, 23, and 24. Specific review criteria are contained in SRP Sections 7.0, 7.2, 7.3, 7.4, 7.7, and 7.8.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's application related to the effects of the proposed EPU on the functional design of the reactor trip system, ESFAS, safe shutdown system, and control systems. The NRC staff concludes that the licensee has adequately addressed the effects of the proposed EPU on these systems and that the changes that are required to achieve the proposed EPU are consistent with the plant's licensing basis. The NRC staff further concludes that the systems will continue to meet the requirements of 10 CFR 50.55a(a)(1), 10 CFR 50.55(a)(h), and GDCs 1, 4, 13, 19, 20, 21, 22, 23, and 24. Therefore, the NRC staff finds the licensee's proposed EPU acceptable with respect to instrumentation and controls.

[2.4.2 Additional Review Areas (Instrumentation and Controls)]

[Insert Regulatory Evaluation, Technical Evaluation, and Conclusion sections as necessary]

INSERT 5

FOR

SECTION 3.3 - PWR TEMPLATE SAFETY EVALUATION

2.5 Plant Systems

2.5.1 Internal Hazards

2.5.1.1 Flooding

2.5.1.1.1 Flood Protection

Regulatory Evaluation

The NRC staff conducts its review in the area of flood protection to ensure that SSCs important to safety are protected from flooding. The NRC staff's review covers flooding of SSCs important to safety from internal sources, such as those caused by failures of tanks and vessels. The NRC staff's review focuses on increases of fluid volumes in tanks and vessels assumed in flooding analyses to assess the impact of any additional fluid on the flooding protection that is provided. The NRC's acceptance criteria for flood protection are based on GDC-2. Specific review criteria are contained in SRP Section 3.4.1.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the proposed changes in fluid volumes in tanks and vessels for the proposed EPU. The NRC staff concludes that SSCs important to safety will continue to be protected from flooding and will continue to meet the requirements of GDC-2 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to flood protection.

2.5.1.1.2 Equipment and Floor Drains

Regulatory Evaluation

The function of the equipment and floor drainage system (EFDS) is to assure that waste liquids, valve and pump leakoffs, and tank drains are directed to the proper area for processing or disposal. The EFDS is designed to handle the volume of leakage expected, prevent a backflow of water that might result from maximum flood levels to areas of the plant containing safety-related equipment, and protect against the potential for inadvertent transfer of contaminated fluids to a non-contaminated drainage system. The NRC staff's review of the EFDS includes the collection and disposal of liquid effluents outside containment. The NRC staff's review is focused on any changes in fluid volumes or pump capacities that are required for the proposed EPU and are not consistent with previous assumptions with respect to floor drainage considerations. The NRC's acceptance criteria for the EFDS are based on GDCs 2 and 4 for the capability of the EFDS to withstand the effects of earthquakes and to be compatible with the environmental conditions (flooding) associated with normal operation, maintenance, testing, and postulated accidents (pipe failures and tank ruptures). Specific review criteria are contained in SRP Section 9.3.3.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the EFDS and concludes that the licensee has adequately accounted for the plant changes resulting in increased water volumes and larger capacity pumps or piping systems. The NRC staff concludes that the EFDS has sufficient capacity to (1) handle the additional expected leakage resulting from the plant changes, (2) prevent the backflow of water to areas with safety-related equipment, and (3) ensure that contaminated fluids are not transferred to noncontaminated drainage systems. Based on this, the NRC staff concludes that the EFDS will continue to meet the requirements of GDCs 2 and 4 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the EFDS.

2.5.1.1.3 Circulating Water System

Regulatory Evaluation

The circulating water system (CWS) provides a continuous supply of cooling water to the main condenser to remove the heat rejected by the turbine cycle and auxiliary systems. The NRC staff's review of the CWS focuses on changes in flooding analyses that are necessary due to increases in fluid volumes or installation of larger capacity pumps or piping required to accommodate the proposed EPU. The NRC's acceptance criteria for the CWS are based on GDC-4 for the effects of flooding of safety-related areas due to leakage from the CWS and the effects of malfunction or failure of a component or piping of the CWS on the functional performance capabilities of safety-related SSCs. Specific review criteria are contained in SRP Section 10.4.5.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the modifications to the CWS and concludes that the licensee has adequately evaluated these modifications. The NRC staff concludes that, consistent with the requirements of GDC-4, the increased volumes of fluid leakage that could potentially result from these modifications would not result in the failure of safety-related SSCs following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the CWS.

2.5.1.2 Missile Protection

2.5.1.2.1. Internally Generated Missiles

Regulatory Evaluation

The NRC staff's review concerns missiles that could result from in-plant component overspeed failures and high-pressure system ruptures. The NRC staff's review of potential missile sources covers pressurized components and systems, and high speed rotating machinery. The NRC staff's review is conducted to ensure that safety related SSCs are adequately protected from internally generated missiles. In addition, if safety-related SSCs are located in areas containing non-safety-related SSCs, the NRC staff reviews the non-safety-related SSCs to ensure that their failure will not preclude the intended safety function of the safety-related SSCs. The NRC staff's review focuses on any increases in system pressures or component overspeed conditions that could result during plant operation, anticipated operational occurrences, or changes in existing system configurations such that missile barrier considerations could be affected. The NRC's acceptance criteria for the protection of SSCs important to safety against the effects of internally generated missiles that may result from equipment failures are based on GDC-4. Specific review criteria are contained in SRP Sections 3.5.1.1 and 3.5.1.2.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the changes in system pressures and configurations that are required for the proposed EPU and concludes that SSCs important to safety will continue to be protected from internally generated missiles and will continue to meet the requirements of GDC-4 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to internally generated missiles.

2.5.1.2.2 Turbine Generator

Regulatory Evaluation

The turbine control system, steam inlet stop and control valves, low pressure turbine steam intercept and inlet control valves, and extraction steam control valves control the speed of the turbine under normal and abnormal conditions, and are thus related to the overall safe operation of the plant. The NRC staff's review of the turbine generator focuses on the effects of the proposed EPU on the turbine overspeed protection features to ensure that a turbine overspeed condition above the design overspeed is very unlikely. The NRC's acceptance criteria for the turbine generator are based on GDC-4 for protection of SSCs important to safety from the effects of turbine missiles by providing a turbine overspeed protection system (with suitable redundancy) to minimize the probability of generating turbine missiles. Specific review criteria are contained in SRP Section 10.2.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the turbine generator and concludes that the licensee has adequately accounted for the effects of changes in plant conditions on turbine overspeed. The NRC staff concludes that the turbine generator will continue to provide adequate turbine overspeed protection to minimize the probability of generating turbine missiles and will continue to meet the requirements of GDC-4 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the turbine generator.

2.5.1.3 Pipe Failures

Regulatory Evaluation

The NRC staff conducts a review of the plant design for protection from piping failures outside containment to ensure that (1) such failures would not cause the loss of needed functions of safety-related systems and (2) the plant could be safely shut down in the event of such failures. The NRC staff's review of pipe failures includes high and moderate energy fluid system piping located outside of containment. The NRC staff's review focuses on the effects of pipe failures on the resulting environmental conditions, control room habitability, and access to areas important to safe control of post-accident operations where the consequences are not bounded by previous analyses. The NRC's acceptance criteria for pipe failures are based on GDC-4 for SSCs important to safety being designed to accommodate the dynamic effects of postulated pipe ruptures, including the effects of pipe whipping and discharging fluids. Specific review criteria are contained in SRP Section 3.6.1.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the changes that are required for the proposed EPU and the licensee's proposed operation of the plant, and concludes that SSCs important to safety will continue to be protected from postulated piping failures in fluid systems outside containment and will continue to meet the requirements of GDC-4 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to protection against postulated piping failures in fluid systems outside containment.

2.5.1.4 Fire Protection

Regulatory Evaluation

The purpose of the fire protection program (FPP) is to provide assurance, through a defense-in-depth design, that a fire will not prevent the performance of necessary safe plant shutdown functions and will not significantly increase the risk of radioactive releases to the environment. The NRC staff's review focuses on the effects of the increased decay heat on the plant's safe shutdown analysis to ensure that SSCs required for the safe shutdown of the plant are protected from the effects of the fire and continue to be able to achieve and maintain safe shutdown following a fire. The NRC's acceptance criteria for the FPP are based on (1) 10 CFR 50.48 and associated Appendix R to 10 CFR Part 50 for the development of a fire protection plan to ensure the capability to safely shut down the plant; (2) GDC-3 for fire prevention, the design and operation of fire detection and suppression systems, and administrative controls provided to protect SSCs important to safety; and (3) GDC-5 for fire protection for shared safety-related SSCs to assure the ability to perform their intended safety function. Specific review criteria are contained in SRP Section 9.5.1, as supplemented by the guidance contained in Attachment 3 to Matrix 2.1 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's fire related safe shutdown assessment and concludes that the licensee has adequately accounted for the effects of the increased decay heat on the ability of the required systems to achieve and maintain safe shutdown conditions. The NRC staff further concludes that the FPP will continue to meet the requirements of 10 CFR 50.48, 10 CFR Part 50, Appendix R, and GDCs 3 and 5 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to fire protection.

2.5.2 Pressurizer Relief Tank

Regulatory Evaluation

The pressurizer relief tank (PRT) is a pressure vessel provided to condense and cool the discharge from the pressurizer safety and relief valves. The capacity of the tank is based on a requirement to absorb discharge fluid from the pressurizer relief valve during a specified step-load decrease. The PRT system is not safety-related and is not designed to accept a continuous discharge from the pressurizer. The NRC staff conducts a review of the PRT to ensure that operation of the tank is consistent with transient analyses of related systems at the proposed EPU level, and that failure or malfunction of the PRT system will not adversely affect safety-related SSCs. The NRC staff's review is focused on any design changes related to the PRT and connected piping, and changes related to operational assumptions that are necessary in support of the proposed EPU that are not bounded by previous analyses. In general, the steam condensing capacity of the tank must be adequate and the tank rupture disk relief capacity must be adequate compared to the capacity of the pressurizer power-operated relief and safety valves, the piping to the tank must be adequately sized, and systems inside containment must be adequately protected from the effects of high-energy line breaks and moderate energy line cracks in the pressurizer relief system. The NRC's acceptance criteria for the PRT are based on GDCs 2 and 4 for the protection of systems from the effects of earthquakes, missiles, or adverse environmental conditions that could result in unnecessary damage to safety-related SSCs. Specific review criteria are contained in SRP Section 5.4.11.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the increase in pressurizer discharge to the PRT as a result of the proposed EPU and concludes that (1) the PRT will operate in a manner consistent with transient analyses of related systems and (2) safety-related SSCs will continue to be protected against failure of the PRT consistent with GDCs 2 and 4. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the design of the PRT.

2.5.3 Fission Product Control

2.5.3.1 Fission Product Control Systems and Structures

Regulatory Evaluation

The NRC staff's review for fission product control systems and structures covers the basis for developing the mathematical model for design-basis LOCA dose computations, the values of key parameters, the applicability of important modeling assumptions, and the functional capability of ventilation systems used to control fission product releases. The NRC staff's review primarily focuses on any adverse effects that the proposed EPU may have on the assumptions used in the analyses for control of fission products. The NRC's acceptance criteria for fission product control systems and structures are based on GDC-41 for the containment atmosphere cleanup system being designed to control fission product releases to the environment following postulated accidents. Specific review criteria are contained in SRP Section 6.5.3.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on fission product control systems and structures. The NRC staff concludes that the licensee has adequately accounted for the increase of fission products and changes in expected environmental conditions that would result from the proposed EPU, and the NRC staff further concludes that the fission product control systems and structures will continue to provide adequate fission product removal in postaccident environments following implementation of the proposed EPU. Based on this, the NRC staff also concludes that the fission product control systems and structures will continue to meet the requirements of GDC-41. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the fission product control systems and structures.

2.5.3.2 Main Condenser Evacuation System

Regulatory Evaluation

The main condenser evacuation system (MCES) generally consists of two subsystems: (1) the "hogging" or startup system which initially establishes main condenser vacuum and (2) the system which maintains condenser vacuum once it has been established. The NRC staff's review focuses on modifications to the system that may affect gaseous radioactive material handling and release assumptions, and design features to preclude the possibility of an explosion (if the potential for explosive mixtures exists). The NRC's acceptance criteria for the MCES are based on (1) GDC-60 for the MCES design for the control of releases of radioactive materials to the environment and (2) GDC-64 for the MCES design for the monitoring of releases of radioactive materials to the environment. Specific review criteria are contained in SRP Section 10.4.2.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment of required changes to the MCES and concludes that the licensee has adequately evaluated these changes. The NRC staff concludes that the MCES will continue to maintain its ability to control and provide monitoring for releases of radioactive materials to the environment following implementation of the proposed EPU. The NRC also concludes that the MCES will continue to meet the requirements of GDCs 60 and 64. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the MCES.

2.5.3.3 Turbine Gland Sealing System

Regulatory Evaluation

The turbine gland sealing system is provided to control the release of radioactive material from steam in the turbine to the environment. The NRC staff reviews changes to the turbine gland sealing system with respect to factors that may affect gaseous radioactive material handling (e.g., source of sealing steam, system interfaces, and potential leakage paths). The NRC's acceptance criteria for the turbine gland sealing system are based on (1) GDC-60 for the turbine gland sealing system design for the control of releases of radioactive materials to the environment and (2) GDC-64 for the turbine gland sealing system design for the monitoring of releases of radioactive materials to the environment. Specific review criteria are contained in SRP Section 10.4.3.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment of required changes to the turbine gland sealing system and concludes that the licensee has adequately evaluated these changes. The NRC staff concludes that the turbine gland sealing system will continue to maintain its ability to control and provide monitoring for releases of radioactive materials to the environment consistent with GDCs 60 and 64. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the turbine gland sealing system.

2.5.4 Component Cooling and Decay Heat Removal

2.5.4.1 Spent Fuel Pool Cooling and Cleanup System

Regulatory Evaluation

The spent fuel pool provides wet storage of spent fuel assemblies. The safety function of the spent fuel pool cooling and cleanup system is to cool the spent fuel assemblies and keep the spent fuel assemblies covered with water during all storage conditions. The NRC staff's review for EPUs focuses on the effects of the proposed EPU on the capability of the system to provide adequate cooling to the spent fuel during all operating and accident conditions. The NRC's acceptance criteria for the spent fuel pool cooling and cleanup system are based on (1) GDC-5 for shared systems and components important to safety being capable of performing required safety functions; (2) GDC-44 for the capability to transfer heat loads from safety-related SSCs to a heat sink under both normal operating and accident conditions; and (3) GDC-61 for RHR capability and measures to prevent a significant loss of fuel storage coolant inventory under accident conditions. Specific review criteria are contained in SRP Section 9.1.3, as supplemented by the guidance contained in Attachment 2 to Matrix 5 of Section 2.1 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment related to the spent fuel pool cooling and cleanup system and concludes that the licensee has adequately accounted for the effects of the proposed EPU on the spent fuel pool cooling and cleanup function of the system. Based on this review, the NRC staff concludes that the spent fuel pool cooling and cleanup system will continue to provide sufficient cooling capability to cool the spent fuel pool following implementation of the proposed EPU and will continue to meet the requirements of GDCs 5, 44, and 61. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the spent fuel pool cooling and cleanup system.

2.5.4.2 Station Service Water System

Regulatory Evaluation

The station service water system (SWS) provides essential cooling to safety-related equipment and may also provide cooling to non-safety-related auxiliary components that are used for normal plant operation. The NRC staff's review covers the characteristics of the station SWS components with respect to their functional performance as affected by adverse operational (i.e., water hammer) requirements, abnormal operational requirements, and accident conditions (e.g., LOCA with the loss of offsite power (LOOP)). The NRC staff's review focuses on the additional heat load that results from the proposed EPU. The NRC's acceptance criteria for the SWS are based on (1) GDC-4 for dynamic effects associated with flow instabilities and loads (e.g., water hammer) during normal plant operation as well as during upset or accident conditions; (2) GDC-5 for the capability of shared systems and components important to safety to perform their required safety functions; and (3) GDC-44 for transferring heat from SSCs important to safety to an ultimate heat sink. Specific review criteria are contained in SRP Section 9.2.1, as supplemented by GL 89-13 and GL 96-06.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the station SWS and concludes that the licensee has adequately accounted for the increased heat loads from the proposed EPU on system performance. The NRC staff concludes that the station SWS will continue to be protected from the dynamic effects associated with flow instabilities and provide sufficient cooling for SSCs important to safety following implementation of the proposed EPU. Therefore, the NRC staff has determined that the station SWS will continue to meet the requirements of GDCs 4, 5, and 44. Based on the above, the NRC staff finds the proposed EPU acceptable with respect to the station SWS.

2.5.4.3 Reactor Auxiliary Cooling Water Systems

Regulatory Evaluation

The NRC staff's review covers reactor auxiliary cooling water systems (CWS) that are required for (1) safe shutdown during normal operations, anticipated operational occurrences, and for mitigating the consequences of accident conditions, or (2) preventing the occurrence of an accident. These systems include closed loop auxiliary cooling water systems for reactor system components, reactor shutdown equipment, ventilation equipment, and components of the ECCS. The NRC staff's review covers the capability of the auxiliary cooling water systems to provide adequate cooling water to safety-related ECCS components and reactor auxiliary equipment for all planned operating conditions. Emphasis is placed on the CWS for safety-related components such as ECCS equipment, ventilation equipment, and reactor shutdown equipment. The NRC staff's review focuses on the additional heat load that results from the proposed EPU. The NRC's acceptance criteria for the reactor auxiliary CWS are based on (1) GDC-4 for dynamic effects associated with flow instabilities and attendant loads (i.e., water hammer) during normal plant operation as well as during upset or accident conditions; (2) GDC-5 for shared systems and components important to safety being capable of performing required safety functions; and (3) GDC-44 for the capability to transfer heat loads from safety-related SSCs to a heat sink under both normal operating and accident conditions. Specific review criteria are contained in SRP Section 9.2.2, as supplemented by GL 89-13 and GL 96-06.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the reactor auxiliary CWS and concludes that the licensee has adequately accounted for the increased heat loads from the proposed EPU on system performance. The NRC staff concludes that the reactor auxiliary CWS will continue to be protected from the dynamic effects associated with flow instabilities and provide sufficient cooling for SSCs important to safety following implementation of the proposed EPU. Therefore the NRC staff has determined that the reactor auxiliary CWS will continue to meet the requirements of GDCs 4, 5, and 44. Based on the above, the NRC staff finds the proposed EPU acceptable with respect to the reactor auxiliary CWS.

2.5.4.4 Ultimate Heat Sink

Regulatory Evaluation

The ultimate heat sink (UHS) is the source of cooling water provided to dissipate reactor decay heat and essential cooling system heat loads after a normal reactor shutdown or a shutdown following an accident. The NRC staff's review is focused on the impact that the proposed EPU has on the decay heat removal capability of the UHS. Additionally, the NRC staff's review includes evaluation of the design-basis UHS temperature limit determination to confirm that post-licensing data trends (e.g., air and water temperatures, humidity, wind speed, water volume) do not establish more severe conditions than previously assumed. The NRC's acceptance criteria for the UHS are based on (1) GDC-5 for shared systems and components important to safety being capable of performing required safety functions, and (2) GDC-44 for the capability to transfer heat loads from safety-related SSCs to the heat sink under both normal operating and accident conditions. Specific review criteria are contained in SRP Section 9.2.5.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the information that was provided by the licensee for addressing the effects that the proposed EPU would have on the UHS safety function, including the licensee's validation of the design-basis UHS temperature limit based on post-licensing data. Based on the information that was provided, the NRC staff concludes that the proposed EPU will not compromise the design-basis safety function of the UHS, and that the UHS will continue to satisfy the requirements of GDCs 5 and 44. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the UHS.

2.5.4.5 Auxiliary Feedwater System

Regulatory Evaluation

In conjunction with a seismic Category I water source, the auxiliary feedwater system (AFWS), functions as an emergency system for the removal of heat from the primary system when the main feedwater system is not available. The AFWS may also be used to provide decay heat removal necessary for withstanding or coping with a station blackout. The NRC staff's review for the proposed EPU focuses on the system's continued ability to provide sufficient emergency feedwater flow at the expected conditions (e.g, steam generator pressure) to ensure adequate cooling with the increased decay heat. The NRC staff's review also considers the effects of the proposed EPU on the likelihood of creating fluid flow instabilities (e.g., waterhammer) during normal plant operation, as well as during upset or accident conditions. The NRC's acceptance criteria for the AFWS are based on (1) GDC-4 for the system itself being capable of withstanding the dynamic effects associated with possible fluid flow instabilities (e.g., waterhammer), and the effects of internally generated missiles, pipe whip, and jet impingement forces associated with pipe breaks; (2) GDC-5 for the capability of shared systems and components important to safety to perform required safety functions; (3) GDC-19 for the design of system instrumentation and controls required for prompt hot shutdown of the reactor and for subsequent cold shutdown; and (4) GDCs 34 and 44 for the capability to transfer heat loads from the reactor system to a heat sink under both normal operating and accident conditions, and the capability to isolate components, subsystems, or piping if required so that the system safety function will be maintained. Specific review criteria are contained in SRP Section 10.4.9.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment related to the AFWS. The NRC staff concludes that the licensee has adequately accounted for the effects of the increase in decay heat and other changes in plant conditions on the ability of the AFWS to supply adequate water to the steam generators to ensure adequate cooling of the core. The NRC staff finds that the AFWS will continue meet its design functions following implementation of the proposed EPU. The NRC staff further concludes that the AFWS will continue to meet the requirements of GDCs 4, 5, 19, 34, and 44. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the AFWS.

2.5.5 Balance-of-Plant Systems

2.5.5.1 Main Steam

Regulatory Evaluation

The main steam supply system (MSSS) transports steam from the nuclear steam supply system to the power conversion system and various safety-related and non-safety-related auxiliaries. Portions of the MSSS may be used as a part of the heat sink to remove heat from the reactor facility during certain operations. The MSSS may also include provisions for secondary system pressure relief. The NRC staff's review focuses on the effects of the proposed EPU on the system's capability to transport steam to the power conversion system, provide heat sink capacity and pressure relief capability, supply steam to drive safety system pumps, and withstand adverse dynamic loads (e.g., water steam hammer resulting from rapid valve closure and relief valve fluid discharge loads). The NRC staff's review also covers the measures provided to limit blowdown of the system in the event of a steamline break. The NRC's acceptance criteria for the MSSS are based on (1) GDC-4 for safety-related portions of the system being capable of withstanding the effects of external missiles and internally generated missiles, pipe whip, and jet impingement forces associated with pipe breaks; (2) GDC-5 for the capability of shared systems and components important to safety to perform required safety functions; and (3) GDC-34 for the system function of transferring residual and sensible heat from the reactor system. Specific review criteria are contained in SRP Section 10.3.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the MSSS and concludes that the licensee has adequately accounted for the effects of changes in plant conditions on the design of the MSSS. The NRC staff concludes that the MSSS will continue to maintain its ability to transport steam to the power conversion system, provide heat sink capacity and pressure relief capability, supply steam to steam-driven safety pumps, and withstand steam hammer. The NRC staff further concludes that the MSSS will continue to meet the requirements of GDCs 4, 5, and 34. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the MSSS.

2.5.5.2 Main Condenser

Regulatory Evaluation

The main condenser (MC) system is designed to condense and deaerate the exhaust steam from the main turbine and provide a heat sink for the turbine bypass system. The NRC staff's review focuses on the effects of the proposed EPU on the steam bypass capability with respect to load rejection assumptions, and on the ability of the MC system to withstand the blowdown effects of steam from the turbine bypass system. The NRC's acceptance criteria for the MC system are based on GDC-60 such that failures in the design of the system are not allowed to result in excessive releases of radioactivity to the environment or in unacceptable condensate quality, or in flooding of areas housing safety-related equipment. Specific review criteria are contained in SRP Section 10.4.1.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the MC system and concludes that the licensee has adequately accounted for the effects of changes in plant conditions on the design of the MC system. The NRC staff concludes that the MC system will continue to maintain its ability to withstand the blowdown effects of the steam from the turbine bypass system and thereby continue to meet GDC-60 for prevention of the consequences of failures in the system. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the MC system.

2.5.5.3 Turbine Bypass

Regulatory Evaluation

The turbine bypass system (TBS) is designed to discharge a stated percentage of rated main steam flow directly to the MC system, bypassing the turbine. This steam bypass enables the plant to take step load reductions up to the TBS capacity without the reactor or turbine tripping. The system is also used during startup and shutdown to control steam generator pressure. The NRC staff's review focuses on the effects that EPU has on load rejection capability, analysis of postulated system piping failures, and on the consequences of inadvertent TBS operation. The NRC's acceptance criteria for the TBS are based on (1) GDC-4 for failure of the TBS due to a pipe break or malfunction of the TBS not adversely affecting essential systems or components; and (2) GDC-34 for the ability to use the system for shutting down the plant during normal operations. Specific review criteria are contained in SRP Section 10.4.4.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the TBS. The NRC staff concludes that the licensee has adequately accounted for the effects of changes in plant conditions on the design of the system. The NRC staff concludes that the TBS will continue to provide a means for shutting down the plant during normal operations. The NRC staff further concludes that TBS failures will not adversely affect essential systems or components. Based on this, the NRC staff concludes that the TBS will continue to meet GDCs 4 and 34. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the TBS.

2.5.5.4 Condensate and Feedwater

Regulatory Evaluation

The condensate and feedwater system (CFS) provides feedwater at the required temperature, pressure, and flow rate to the steam generators. The only part of the CFS classified as safety-related is the feedwater piping from the steam generators up to and including the outermost containment isolation valve. The NRC staff's review focuses on the effects of the proposed EPU on previous analyses and considerations with respect to the capability of the CFS to supply adequate feedwater during plant operation and shutdown, and to isolate components, subsystems, and piping in order to preserve the system safety function. The NRC staff's review also considers the effects of EPU on the feedwater system, including the auxiliary feedwater system piping entering the steam generator, with regard to possible fluid flow instabilities (e.g., water hammer) during normal plant operation as well as during upset or accident conditions. The NRC's acceptance criteria for the CFS are based on (1) GDC-4 for the dynamic effects associated with possible fluid flow instabilities (e.g., water hammers) during normal plant operation as well as during upset or accident conditions; (2) GDC-5 for the capability of shared systems and components important to safety to perform required safety functions; and (3) GDC-44 for satisfying feedwater flow requirements and system isolation considerations. Specific review criteria are contained in SRP Section 10.4.7.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the CFS and concludes that the licensee has adequately accounted for the effects of changes in plant conditions on the design of the CFS. The NRC staff concludes that the CFS will continue to maintain its ability to satisfy feedwater requirements for normal operation and shutdown, withstand water hammer, maintain isolation capability in order to preserve the system safety function, and not cause failure of safety-related SSCs. The NRC staff further concludes that the CFS will continue to meet the requirements of GDCs 4, 5, and 44. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the CFS.

2.5.6 Waste Management Systems

2.5.6.1 Gaseous Waste Management Systems

Regulatory Evaluation

Gaseous waste management systems involve the gaseous radwaste system, which deals with the management of radioactive gases collected in the offgas system or the waste gas storage and decay tanks. In addition, it involves the management of condenser air removal system, steam generator blowdown flash tank, and containment purge exhausts; and building ventilation system exhausts. The NRC staff's review is focused on the effects that EPU has on previous analyses and considerations related to the gaseous waste management systems' design criteria, methods of treatment, expected releases, principal parameters used in calculating the releases of radioactive materials in gaseous effluents, and design features to preclude the possibility of an explosion if the potential for explosive mixtures exist. The NRC's acceptance criteria for the gaseous waste management systems are based on (1) 10 CFR 20.1302 for radioactivity in effluents; (2) GDC-3 for providing protection for gaseous waste handling and treatment systems from the effects of an explosive mixture of hydrogen and oxygen; (3) GDC-60 for designing gaseous waste management systems to control releases of radioactive materials to the environment; (4) GDC-61 for radioactivity control in gaseous waste management systems associated with fuel storage and handling areas; and (5) 10 CFR Part 50 Appendix I, Sections II.B., II.C., and II.D., for the numerical guides for design objectives and limiting conditions for operation to meet the "as low as is reasonably achievable" criterion. Specific review criteria are contained in SRP Section 11.3.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment related to the gaseous waste management systems. The NRC staff concludes that the licensee has adequately accounted for the effects of the increase in fission product and amount of gaseous waste on the abilities of the systems to control releases of radioactive materials and preclude the possibility of an explosion if the potential for explosive mixtures exists. The NRC staff finds that the gaseous waste management systems will continue to meet their design functions following implementation of the proposed EPU. The NRC staff further concludes that the gaseous waste management systems will continue to meet the requirements of 10 CFR 20.1302, GDCs 3, 60, and 61, and 10 CFR 50, Appendix I, Sections II.B, II.C, and II.D. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the gaseous waste management systems.

2.5.6.2 Liquid Waste Management Systems

Regulatory Evaluation

The NRC staff's review for liquid waste management systems is focused on the effects that the proposed EPU have on previous analyses and considerations related to the liquid waste management systems' design, design objectives, design criteria, methods of treatment, expected releases, and principal parameters used in calculating the releases of radioactive materials in liquid effluents. The NRC's acceptance criteria for the liquid waste management systems are based on (1) 10 CFR 20.1302 for radioactivity in effluents to unrestricted areas; (2) GDC-60 for the liquid waste management systems being designed to control releases of radioactive materials to the environment; (3) GDC-61 for the liquid waste management systems being designed to ensure adequate safety under normal and postulated accident conditions; and (4) 10 CFR Part 50, Appendix I, Sections II.A and II.D for the numerical guides for dose design objectives and limiting conditions for operation to meet the "as low as is reasonably achievable" criterion. Specific review criteria are contained in SRP Section 11.2.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment related to the liquid waste management systems. The NRC staff concludes that the licensee has adequately accounted for the effects of the increase in fission product and amount of liquid waste on the ability of the liquid waste management systems to control releases of radioactive materials. The NRC staff finds that the liquid waste management systems will continue to meet their design functions following implementation of the proposed EPU. The NRC staff further concludes that the licensee has demonstrated that the liquid waste management systems will continue to meet the requirements of 10 CFR 20.1302, GDCs 60 and 61, and 10 CFR Part 50, Appendix I, Sections II.A and II.D. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the liquid waste management systems.

2.5.6.3 Solid Waste Management Systems

Regulatory Evaluation

The NRC staff's review is focused on the effects that the proposed EPU have on previous analyses and considerations related to the design objectives in terms of expected volumes of waste to be processed and handled, the wet and dry types of waste to be processed, the activity and expected radionuclide distribution contained in the waste, equipment design capacities, and the principal parameters employed in the design of the solid waste management systems (SWMS). The NRC's acceptance criteria for the SWMS are based on (1) 10 CFR 20.1302 for radioactive materials released in gaseous and liquid effluents to unrestricted areas; (2) GDC-60 for the SWMS being designed with means to handle solid wastes produced during normal plant operation, including operational occurrences; (3) GDC-63 and 64 for the radioactive waste system being designed for monitoring radiation levels and leakage; and (4) 10 CFR Part 71 for radioactive material packaging. Specific review criteria are contained in SRP Section 11.4.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment related to the SWMS. The NRC staff concludes that the licensee has adequately accounted for the effects of the increase in fission product and amount of solid waste on the ability of the SWMS to process the waste. The NRC staff finds that the SWMS will continue to meet its design functions following implementation of the proposed EPU. The NRC staff further concludes that the licensee has provided sufficient information consistent with 10 CFR 50.34a to demonstrate that the SWMS will continue to meet the requirements of 10 CFR 20.1302, GDCs 60, 63, and 64, and 10 CFR Part 71. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the SWMS.

2.5.7 Additional Considerations

2.5.7.1 Emergency Diesel Engine Fuel Oil Storage and Transfer System

Regulatory Evaluation

Nuclear power plants are required to have redundant onsite emergency power sources of sufficient capacity to power safety-related equipment (e.g., diesel engine-driven generator sets). This section of the safety evaluation deals with the fuel oil storage and transfer system for these diesel engines. The NRC staff's review focuses on increases in emergency diesel generator electrical demand and the resulting increase in required fuel oil. The NRC's acceptance criteria for the emergency diesel engine fuel oil storage and transfer system are based on (1) GDC-4 for the capability to withstand internally generated missiles, pipe whip, and jet impingement forces associated with pipe breaks; (2) GDC-5 for the capability of shared systems and components important to safety to perform required safety functions; and (3) GDC-17 for the capability of the fuel oil system to meet independence and redundancy criteria. Specific review criteria are contained in SRP Section 9.5.4.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment related to the amount of required fuel oil for the emergency diesel generators and concludes that the licensee has adequately accounted for the effects of the increased electrical demand on fuel oil consumption. The NRC staff concludes that the fuel oil storage and transfer system will continue to provide an adequate amount of fuel oil to allow the diesel generators to meet the onsite power requirements of GDCs 4, 5, and 17. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the emergency diesel engine fuel oil storage and transfer system.

2.5.7.2 Light Load Handling System (Related to Refueling)

Regulatory Evaluation

The light load handling system (LLHS) includes components and equipment used in handling new fuel at the receiving station to the loading of the spent fuel into the shipping cask. The NRC staff's review covers the avoidance of criticality accidents, radioactivity releases resulting from damage to irradiated fuel, and unacceptable personnel radiation exposures. The NRC staff's review is focused on the effects of the new fuel on system performance and related analyses. The NRC's acceptance criteria for the LLHS are based on (1) GDC-61 for radioactivity release as a result of fuel damage, and the avoidance of excessive personnel radiation exposure; and (2) GDC-62 for criticality accidents. Specific review criteria are contained in SRP Section 9.1.4.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment related to the effects of the new fuel on the ability of the LLHS to avoid criticality accidents and concludes that the licensee has adequately incorporated the effects of the new fuel in the analyses. Based on this review, the NRC staff further concludes that the LLHS will continue to meet the requirements of GDCs 61 and 62 for radioactivity releases and prevention of criticality accidents. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the LLHS.

[2.5.8 Additional Review Areas (Plant Systems)]

[Insert Regulatory Evaluation, Technical Evaluation, and Conclusion sections as necessary]

INSERT 6

FOR

SECTION 3.3 - PWR TEMPLATE SAFETY EVALUATION

2.6 Containment Review Considerations

2.6.1 Primary Containment Functional Design

Regulatory Evaluation

The containment encloses the reactor system and is the final barrier against the release of significant amounts of radioactive fission products in the event of an accident. The containment structure must be capable of withstanding, without loss of function, the pressure and temperature conditions resulting from postulated LOCAs, steamline accidents, or feedwater-line-break accidents. The containment structure must continue to function as a low leakage barrier against the release of fission products for as long as postulated accident conditions require.

NOTE: Use the following paragraph in the regulatory evaluation and the conclusion section provided below for Dry Containments, Including Subatmospheric Containments

The NRC staff's review covers the pressure and temperature conditions in the containment due to a spectrum of postulated LOCAs and secondary system line-breaks. The NRC's acceptance criteria for primary containment functional design are based on (1) GDCs 16 and 50 for the containment and its associated systems being able to accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any LOCA; (2) GDC-38 for the containment heat removal system(s) function to rapidly reduce the containment pressure and temperature following any LOCA and maintain them at acceptably low levels; (3) GDC-13 for instrumentation to monitor variables and systems over their anticipated ranges for normal operation and for accident conditions; and (4) GDC-64 for monitoring the reactor containment atmosphere for radioactivity that may be released from normal operations and postulated accidents. Specific review criteria are contained in SRP Section 6.2.1.1.A.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the containment pressure and temperature transient and concludes that the licensee has adequately accounted for the increase of mass and energy that would result from the proposed EPU. The NRC staff further concludes that containment systems will continue to provide sufficient pressure and temperature mitigation capability to ensure that containment integrity is maintained. The NRC staff also concludes that the containment systems and instrumentation will continue to be adequate for monitoring containment parameters and release of radioactivity during normal and accident conditions and will continue to meet the requirements of GDCs 13, 16, 38, 50, and 64 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to containment functional design.

NOTE: Use the following paragraph in the regulatory evaluation and the conclusion section provided below for Ice Condenser Containments

The NRC staff's review covers the pressure and temperature conditions in the containment due to a spectrum of LOCAs and secondary system line-breaks, the design of the ice condenser system, and the maximum allowable operating deck steam bypass area for a full spectrum of RCS pipe breaks. The NRC's acceptance criteria for primary containment functional design are based on (1) GDCs 16 and 50 for the containment and its associated systems being able to accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any LOCA; (2) GDC-38 for the containment heat removal system(s) function to rapidly reduce the containment pressure and temperature following any LOCA and maintain them at acceptably low levels; (3) GDC-13 for instrumentation to monitor variables and systems over their anticipated ranges for normal operation and for accident conditions, as appropriate, to assure adequate safety; and (4) GDC-64 for monitoring the reactor containment atmosphere for radioactivity that may be released from normal operations and postulated accidents. Specific review criteria are contained in SRP Section 6.2.1.1.B.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the containment pressure and temperature transient and concludes that the licensee has adequately accounted for the increase of mass and energy that would result from the proposed EPU. The NRC staff further concludes that containment systems will continue to provide sufficient pressure and temperature mitigation capability to ensure that containment integrity is maintained. The NRC staff also concludes that containment systems and instrumentation will continue to be adequate for monitoring containment parameters and release of radioactivity during normal and accident conditions and will continue to meet the requirements of GDCs 13, 16, 38, 50, and 64 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to containment functional design.

2.6.2 Subcompartment Analyses

Regulatory Evaluation

A subcompartment is defined as any fully or partially enclosed volume within the primary containment that houses high energy piping and would limit the flow of fluid to the main containment volume in the event of a postulated pipe rupture within the volume. The NRC staff's review for subcompartment analyses covers the determination of the design differential pressure values for containment subcompartments. The NRC staff's review focuses on the effects of the increase in mass and energy release into the containment and the resulting increase in pressurization. The NRC's acceptance criteria for subcompartment analyses are based on (1) GDC-4 for the environmental and missile protection provided to assure that SSCs important to safety are designed to accommodate the dynamic effects that may occur during normal plant operations or during an accident, and (2) GDC-50 for the subcompartments being designed with sufficient margin to prevent fracture of the structure due to pressure differential across the walls of the subcompartment. Specific review criteria are contained in SRP Section 6.2.1.2.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the subcompartment assessment performed by the licensee and the change in predicted pressurization that would result from the increased mass and energy release. The NRC staff concludes that SSCs important to safety will continue to be protected from the dynamic effects that would result from the pipe breaks and that the subcompartments will continue to have sufficient margins to prevent fracture of the structure due to pressure difference across the walls. Based on this review, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 4 and 50 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to subcompartment analyses.

2.6.3 Mass and Energy Release

2.6.3.1. Mass and Energy Release Analysis for Postulated Loss-of-Coolant

Regulatory Evaluation

The release of high-energy fluid into containment from pipe breaks could challenge the structural integrity of the containment, including subcompartments and systems within the containment. The NRC staff's review covers the energy sources that are available for release to the containment and the mass and energy release rate calculations for the initial blowdown, core reflood, and post-reflood phases of the accident. The NRC's acceptance criteria for the mass and energy release analysis for postulated LOCAs are based on (1) GDC-50 for providing sufficient conservatism in the mass and energy release analysis to assure that containment design margin is maintained and (2) 10 CFR Part 50, Appendix K, for sources of energy during the LOCA. Specific review criteria are contained in SRP Section 6.2.1.3.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's mass and energy release assessment and concludes that the licensee has adequately addressed the effects of the proposed EPU and has appropriately accounted for the sources of energy identified in 10 CFR Part 50, Appendix K. Based on this, the NRC staff finds that the mass and energy release analysis meets the requirements in GDC-50 for ensuring that the analysis is conservative. Therefore, the NRC staff finds the proposed EPU acceptable with respect to mass and energy release for postulated LOCA.

2.6.3.2 Mass and Energy Release Analysis for Secondary System Pipe Ruptures

Regulatory Evaluation

The NRC staff's review covers the energy sources that are available for release to the containment, the mass and energy release rate calculations, and the single-failure analyses performed for steam and feedwater line isolation provisions which would limit the flow of steam or feedwater to the assumed pipe rupture. The NRC's acceptance criteria for mass and energy release analysis for secondary system pipe ruptures are based on GDC-50 for providing sufficient conservatism in the mass and energy release analysis for postulated secondary system pipe ruptures to assure that the containment design margin is maintained. Specific review criteria are contained in SRP Section 6.2.1.4.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the mass and energy release assessment performed by the licensee for postulated secondary system pipe ruptures and finds that the licensee has adequately addresses the effects of the proposed EPU. Based on this, the NRC staff concludes that the analysis meets the requirements in GDC-50 for ensuring that the analysis is conservative. Therefore, the NRC staff finds the proposed EPU acceptable with respect to mass and energy release for postulated secondary system pipe ruptures.

2.6.4 Combustible Gas Control in Containment

Regulatory Evaluation

Following a LOCA, hydrogen and oxygen may accumulate inside the containment due to chemical reaction between the fuel rod cladding and steam, corrosion of aluminum and other materials, and radiolytic decomposition of water. If excessive hydrogen is generated, it may form a combustible mixture in the containment atmosphere. The NRC staff's review covers (1) the production and accumulation of the combustible gases, (2) the capability to prevent high concentrations of combustible gases in local areas, (3) the capability to monitor combustible gas concentrations, and (4) the capability to reduce combustible gas concentrations. The NRC staff's review is primarily focused on any impact that the proposed EPU may have on hydrogen release assumptions, and how increases in hydrogen release are mitigated. The NRC's acceptance criteria for combustible gas control in containment are based on (1) 10 CFR 50.44 and 10 CFR 50.46 for plants being designed to prevent the development of combustible mixtures in the containment atmosphere; (2) GDC-5 for shared systems and components important to safety being able to perform required safety functions; and (3) GDCs 41, 42, and 43 for systems being provided to control the concentration of hydrogen or oxygen that may be released into the reactor containment following postulated accidents to ensure that containment integrity is maintained. **[*Include the following sentence for PWRs with ice condenser containments*: Additional requirements based on 10 CFR 50.44 for control of combustible gas during severe accidents apply to plants with deliberate ignition systems.]** Specific review criteria are contained in SRP Section 6.2.5.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment related to combustible gas and concludes that the plant will continue to have sufficient capabilities, consistent with the requirements in 10 CFR 50.44, 10 CFR 50.46, and GDCs 5, 41, 42, and 43 as discussed in the Regulatory Evaluation section above. Therefore, the NRC staff finds the proposed EPU acceptable with respect to combustible gas control in containment.

2.6.5 Containment Heat Removal

Regulatory Evaluation

Fan cooler systems, spray systems, and residual heat removal (RHR) systems are provided to remove heat from the containment atmosphere and from the water in the containment sump. The NRC staff's review in this area focuses on (1) the effects of the proposed EPU on the analyses of the available net positive suction head (NPSH) to the containment heat removal system pumps and (2) the analyses of the heat removal capabilities of the spray water system and the fan cooler heat exchangers. The NRC's acceptance criteria for containment heat removal are based on GDC-38 for the containment heat removal system being capable of rapidly reducing the containment pressure and temperature following a LOCA, and maintaining them at acceptably low levels. Specific review criteria are contained in SRP Section 6.2.2 as supplemented by Draft Guide (DG) 1107.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the containment heat removal systems assessment provided by the licensee and concludes that the licensee has adequately addressed the effects of the proposed EPU. The NRC staff finds that the systems will continue to meet GDC-38 for rapidly reducing the containment pressure and temperature following a LOCA, and maintaining them at acceptably low levels. Therefore, the NRC staff finds the proposed EPU acceptable with respect to containment heat removal systems.

2.6.6 Pressure Analysis for ECCS Performance Capability

Regulatory Evaluation

Following a LOCA, the emergency core cooling system (ECCS) will supply water to the reactor vessel to reflood, and thereby cool the reactor core. The core flooding rate will increase with increasing containment pressure. The NRC staff reviews analyses of the minimum containment pressure that could exist during the period of time until the core is reflooded to confirm the validity of the containment pressure used in ECCS performance capability studies. The NRC staff's review covers assumptions made regarding heat removal systems, structural heat sinks, and other heat removal processes that have the potential to reduce the pressure. The NRC's acceptance criteria for the pressure analysis for ECCS performance capability are based on 10 CFR 50.46 for the use of either an acceptable ECCS evaluation model that realistically describes the behavior of the reactor during LOCAs or an ECCS evaluation model developed in conformance with 10 CFR Part 50, Appendix K. Specific review criteria are contained in SRP Section 6.2.1.5.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the impact that the proposed EPU would have on the minimum containment pressure analysis and concludes that the licensee has adequately addressed this area of review to ensure that the requirements in 10 CFR 50.46 regarding ECCS performance will continue to be met. Therefore, the NRC staff finds the proposed EPU acceptable with respect to minimum containment pressure for ECCS performance.

[2.6.7 Additional Review Areas (Containment Review Considerations)]

[Insert Regulatory Evaluation, Technical Evaluation, and Conclusion sections as necessary]

INSERT 7

FOR

SECTION 3.3 - PWR TEMPLATE SAFETY EVALUATION

2.7 Habitability, Filtration, and Ventilation

2.7.1 Control Room Habitability System

Regulatory Evaluation

The NRC staff reviews the control room habitability system and control building layout and structures to ensure that plant operators are adequately protected from the effects of accidental releases of toxic and radioactive gases. A further objective of the NRC staff's review is to ensure that the control room can be maintained as the backup center from which technical support center personnel can safely operate the plant in the case of an accident. The NRC staff's review focuses on the effects of the proposed EPU on the radiation doses, toxic gas concentrations, and estimates of dispersion of airborne contamination. The NRC's acceptance criteria for the control room habitability system are based on (1) GDC-4 for accommodating the effects of and being compatible with postulated accidents, including the effects of the release of toxic gases; and (2) GDC-19 for maintaining the control room in a safe, habitable condition during accidents by providing adequate protection against radiation and toxic gases. Specific review criteria are contained in SRP Section 6.4 and other guidance provided in Matrix 5 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the ability of the control room habitability system to protect plant operators against the effects of accidental releases of toxic and radioactive gases. The NRC staff concludes that the licensee has adequately accounted for the increase of toxic and radioactive gases that would result from the proposed EPU and the NRC staff further concludes that the control room habitability system will continue to provide the required protection following implementation of the proposed EPU. Based on this, the NRC staff concludes that the control room habitability system will continue to meet the requirements of GDCs 4 and 19. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the control room habitability system.

2.7.2 Engineered Safety Feature Atmosphere Cleanup

Regulatory Evaluation

Engineered safety feature (ESF) atmosphere cleanup systems are designed for fission product removal in postaccident environments. These systems generally include primary systems (e.g., in-containment recirculation) and secondary systems (e.g., emergency or postaccident air-cleaning systems) for the fuel-handling building, control room, shield building, and areas containing ESF components. For each ESF atmosphere cleanup system, the NRC staff's review focuses on the effects of the proposed EPU on system functional design; environmental design; and provisions to inhibit offdesign temperatures in the adsorber section. The NRC's acceptance criteria for the ESF atmosphere cleanup systems are based on (1) GDC-19 for the design of systems for habitability of the control room under accident conditions; (2) GDC-41 for the design of systems to be used for containment atmosphere cleanup following postulated accidents and to control releases to the environment; (3) GDC-61 for the design of systems for radioactivity control under normal and postulated accident conditions; and (4) GDC-64 for monitoring radioactive releases from ESF atmosphere cleanup systems under normal, anticipated operational occurrences, and postulated accident conditions. Specific review criteria are contained in SRP Section 6.5.1.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the ESF atmosphere cleanup systems. The NRC staff concludes that the licensee has adequately accounted for the increase of fission products and changes in expected environmental conditions that would result from the proposed EPU, and the NRC staff further concludes that the ESF atmosphere cleanup systems will continue to provide adequate fission product removal in postaccident environments following implementation of the proposed EPU. Based on this, the NRC staff concludes that the ESF atmosphere cleanup systems will continue to meet the requirements of GDCs 19, 41, 61, and 64. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the ESF atmosphere cleanup systems.

2.7.3 Ventilation Systems

2.7.3.1. Control Room Area Ventilation System

Regulatory Evaluation

The function of the control room area ventilation system (CRAVS) is to provide a controlled environment for the comfort and safety of control room personnel and to support the operability of control room components during normal operation, anticipated operational occurrences, and design basis accident conditions. The NRC staff's review of the CRAVS focuses on the effects that the proposed EPU will have on the functional performance of safety-related portions of the system. The review includes the effects of radiation, combustion, and other toxic products; and the expected environmental conditions in areas served by the CRAVS. The NRC's acceptance criteria for the CRAVS are based on (1) GDC-4 for the CRAVS being designed to accommodate the effects of and to be compatible with anticipated environmental conditions; (2) GDC-19 for providing adequate protection to permit access and occupancy of the control room under accident conditions; and (3) GDC-60 for the system's capability to suitably control release of gaseous radioactive effluents to the environment. Specific review criteria are contained in SRP Section 9.4.1.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the ability of the CRAVS to provide a controlled environment for the comfort and safety of control room personnel and to support the operability of control room components. The NRC staff concludes that the licensee has adequately accounted for the increase of toxic and radioactive gases that would result from the proposed EPU and changes to parameters affecting environmental conditions for control room personnel and equipment. The NRC staff concludes that the CRAVS will continue to provide an acceptable control room environment for safe operation of the plant following implementation of the proposed EPU. The NRC staff also concludes that the system will continue to suitably control the release of gaseous radioactive effluents to the environment. Based on this, the NRC staff concludes that the CRAVS will continue to meet the requirements of GDCs 4, 19 and 60. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the CRAVS.

2.7.4 Spent Fuel Pool Area Ventilation System

Regulatory Evaluation

The function of the spent fuel pool area ventilation system (SFPAVS) is to maintain ventilation in the spent fuel pool equipment areas, to permit personnel access, and to control airborne radioactivity in the area during normal operation, anticipated operational occurrences, and following postulated fuel handling accidents. The NRC staff's review focuses on the effects of the proposed EPU on the functional performance of the safety-related portions of the system. The NRC's acceptance criteria for the SFPAVS are based on (1) GDC-60 for the system's capability to suitably control release of gaseous radioactive effluents to the environment; and (2) GDC-61 for the system's capability to provide appropriate containment. Specific review criteria are contained in SRP Section 9.4.2.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the SFPAVS. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed EPU on the system's capability to maintain ventilation in the spent fuel pool equipment areas, to permit personnel access, control airborne radioactivity in the area, control release of gaseous radioactive effluents to the environment, and provide appropriate containment. Based on this, the NRC staff concludes that the SFPAVS will continue to meet the requirements of GDCs 60 and 61. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the SFPAVS.

2.7.5 Auxiliary and Radwaste Area and Turbine Area Ventilation Systems

Regulatory Evaluation

The function of the auxiliary and radwaste area ventilation system (ARAVS) and the turbine area ventilation system (TAVS) is to maintain ventilation in the auxiliary and radwaste equipment and turbine areas, to permit personnel access, and to control the concentration of airborne radioactive material in these areas during normal operation, during anticipated operational occurrences, and after postulated accidents. The NRC staff's review focuses on the effects of the proposed EPU on the functional performance of the safety-related portions of these systems. The NRC's acceptance criteria for the ARAVS and TAVS are based on GDC-60 for the capability of these systems to suitably control release of gaseous radioactive effluents to the environment. Specific review criteria are contained in SRP Sections 9.4.3 and 9.4.4.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the ARAVS and TAVS. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed EPU on the capability of these systems to maintain ventilation in the auxiliary and radwaste equipment areas and in the turbine area, permit personnel access, control the concentration of airborne radioactive material in these areas, and control release of gaseous radioactive effluents to the environment. Based on this, the NRC staff concludes that the ARAVS and TAVS will continue to meet the requirements of GDC-60. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the ARAVS and the TAVS.

2.7.6 Engineered Safety Feature Ventilation System

Regulatory Evaluation

The function of the engineered safety feature ventilation system (ESFVS) is to provide a suitable and controlled environment for engineered safety feature components following certain anticipated transients and design-basis accidents. The NRC staff's review for the ESFVS focuses on the effects of the proposed EPU on the functional performance of the safety-related portions of the system. The NRC staff's review also covers (1) the ability of the safety features equipment in the areas being serviced by the ventilation system to function under degraded ESFVS system performance; (2) the capability of the ESFVS to circulate sufficient air to prevent accumulation of flammable or explosive gas or fuel-vapor mixtures from components such as storage batteries and stored fuel; (3) and the capability of the system to control airborne particulate material (dust) accumulation. The NRC's acceptance criteria for the ESFVS are based on (1) GDC-4 for the ESFVS being designed to accommodate the effects of and to be compatible with anticipated environmental conditions associated with normal operation and postulated accidents; (2) GDC-17 for ensuring proper functioning of the essential electric power system; and (3) GDC-60 for the system being able to suitably control release of gaseous radioactive effluents to the environment. Specific review criteria are contained in SRP Section 9.4.5.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the ESFVS. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed EPU on the ability of the ESFVS to provide a suitable and controlled environment for ESF components. The NRC staff further concludes that the ESFVS will continue to assure a suitable environment for the ESF components following implementation of the proposed EPU. The NRC staff also concludes that the system will continue to suitably control the release of gaseous radioactive effluents to the environment following implementation of the proposed EPU. Based on this, the NRC staff concludes that the ESFVS will continue to meet the requirements of GDCs 4, 17 and 60. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the ESFVS.

[2.7.7 Additional Review Areas (Habitability, Filtration, and Ventilation)]

[Insert Regulatory Evaluation, Technical Evaluation, and Conclusion sections as necessary]

INSERT 8

FOR

SECTION 3.3 - PWR TEMPLATE SAFETY EVALUATION

2.8 Reactor Systems

2.8.1 Fuel System Design

Regulatory Evaluation

The fuel system consists of arrays of fuel rods, burnable poison rods, spacer grids and springs, end plates, and reactivity control rods. The NRC staff reviews the fuel system to ensure that (1) the fuel system is not damaged as a result of normal operation and anticipated operational occurrences, (2) fuel system damage is never so severe as to prevent control rod insertion when it is required, (3) the number of fuel rod failures is not underestimated for postulated accidents, and (4) coolability is always maintained. The NRC staff's review covers fuel system damage mechanisms, limiting values for important parameters, and performance of the fuel system during normal operation, anticipated operational occurrences, and postulated accidents. The NRC's acceptance criteria are based on (1) 10 CFR 50.46 for core cooling; (2) GDC-10 for assuring that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including anticipated operational occurrences; (3) GDC-27 for the reactivity control system being designed with appropriate margin, and in conjunction with the ECCS, being capable of controlling reactivity and cooling the core under postaccident conditions; and (4) GDC-35 for providing an ECCS to transfer heat from the reactor core following any loss of reactor coolant. Specific review criteria are contained in SRP Section 4.2 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed EPU on the fuel system design. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed EPU on the fuel system and demonstrated that (1) the fuel system will not be damaged as a result of normal operation and anticipated operational occurrences, (2) the fuel system damage will never be so severe as to prevent control rod insertion when it is required, (3) the number of fuel rod failures will not be underestimated for postulated accidents, and (4) coolability will always be maintained. Based on this, the NRC staff concludes that the fuel system and associated analyses will continue to meet the requirements of 10 CFR 50.46, GDC-10, GDC-27, and GDC-35 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the fuel system design.

2.8.2 Nuclear Design

Regulatory Evaluation

The NRC staff reviews the nuclear design of the fuel assemblies, control systems, and reactor core to ensure that fuel design limits will not be exceeded during normal operation or anticipated operational transients, and that the effects of postulated reactivity accidents will not cause significant damage to the RCPB or impair the capability to cool the core. The NRC staff's review covers core power distribution, reactivity coefficients, reactivity control requirements and control provisions, control rod patterns and reactivity worths, criticality, burnup, and vessel irradiation. The NRC's acceptance criteria are based on (1) GDC-10 for assuring that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including anticipated operational occurrences; (2) GDC-11 for the core design to assure that the prompt inherent nuclear feedback characteristics compensate for a rapid increase in reactivity; (3) GDC-12 for precluding or detecting and suppressing power oscillations which could result in conditions exceeding specified acceptable fuel design limits; (4) GDC-13 for instrumentation and controls to monitor variables and systems affecting the fission process over anticipated ranges for normal operation, anticipated operational occurrences and accident conditions, and maintaining the variables and systems within prescribed operating ranges; (5) GDC-20 for automatic initiation of the reactivity control systems to assure that acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and to assure automatic operation of systems and components important to safety under accident conditions; (6) GDC-25 for a single malfunction of the reactivity control system to not cause a violation of the specified acceptable fuel design limits; (7) GDC-26 for providing two independent reactivity control systems of different design, and each system having the capability to control the rate of reactivity changes resulting from planned, normal power changes; (8) GDC-27 for the capability of the reactivity control systems in conjunction with poison addition by the ECCS to reliably control reactivity changes under postulated accident conditions, with appropriate margin for stuck rods; and (9) GDC-28 for the effects of postulated reactivity accidents neither resulting in damage to the RCPB greater than limited local yielding, nor causing sufficient damage to impair significantly the capability to cool the core. Specific review criteria are contained in SRP Section 4.3 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's analyses related to the effect of the proposed EPU on the nuclear design of the fuel assemblies, control systems, and reactor core. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed EPU on the nuclear design and has demonstrated that the fuel design limits will not be exceeded during normal or anticipated operational transients, and that the effects of postulated reactivity accidents will not cause significant damage to the RCPB or impair the capability to cool the core. Based on this evaluation and in coordination with the reviews of the fuel system design, thermal and hydraulic design, and transient and accident analyses, the NRC staff concludes that the nuclear design of the fuel assemblies, control systems, and reactor core will continue to meet the applicable requirements of GDCs 10, 11, 12, 13, 20, 25, 26, 27, and 28. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the nuclear design.

2.8.3 Thermal and Hydraulic Design

Regulatory Evaluation

The NRC staff reviews the thermal and hydraulic design of the core and the RCS to confirm that the design (1) has been accomplished using acceptable analytical methods, (2) is equivalent to or a justified extrapolation from proven designs, (3) provides acceptable margins of safety from conditions which would lead to fuel damage during normal reactor operation and anticipated operational occurrences, and (4) is not susceptible to thermal-hydraulic instability. The review also covers hydraulic loads on the core and RCS components during normal operation and design-basis accident conditions and core thermal-hydraulic stability under conditions of normal operation and anticipated operational occurrences. The NRC's acceptance criteria are based on GDC-10 for the reactor core being designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during normal operation or anticipated operational occurrences. Specific review criteria are contained in SRP Section 4.4 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed EPU on the thermal and hydraulic design of the core and the RCS. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed EPU on the thermal and hydraulic design and demonstrated that the design has been accomplished using acceptable analytical methods, is **[equivalent to or a justified extrapolation from]** proven designs, provides acceptable margins of safety from conditions which would lead to fuel damage during normal reactor operation and anticipated operational occurrences, and is not susceptible to thermal-hydraulic instability. The NRC staff further concludes that the licensee has adequately accounted for the effects of the proposed EPU on the hydraulic loads on the core and RCS components. Based on this, the NRC staff concludes that the thermal and hydraulic design will continue to meet the requirements of GDC-10 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to thermal and hydraulic design.

2.8.4 Emergency Systems

2.8.4.1 Functional Design of Control Rod Drive System

Regulatory Evaluation

The NRC staff's review covers the functional performance of the control rod drive system (CRDS) to confirm that the system can effect a safe shutdown, respond within acceptable limits during anticipated operational occurrences, and prevent or mitigate the consequences of postulated accidents. The review also covers the CRDS cooling system to ensure that it continues to meet its design requirements. The NRC's acceptance criteria are based on (1) GDC-4 for the environmental conditions caused by high or moderate energy pipe breaks during normal plant operation, as well as postulated accidents; (2) GDC-23 for failing into a safe state; (3) GDC-25 for the functional design of redundant reactivity control systems to assure that specified acceptable fuel design limits are not exceeded for malfunction of any reactivity control systems; (4) GDC-26 for the capability of the reactivity control systems to regulate the rate of reactivity changes resulting from normal operations and anticipated operational occurrences; (5) GDC-27 for the combined capability of reactivity control systems and the emergency core cooling system to reliably control reactivity changes to assure the capability to cool the core under accident conditions; (6) GDC-28 for postulated reactivity accidents; and (7) GDC-29 for functioning under anticipated operational occurrences. Specific review criteria are contained in SRP Section 4.6.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed EPU on the functional design of the CRDS. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed EPU on the system and demonstrated that the system's ability to effect a safe shutdown, respond within acceptable limits, and prevent or mitigate the consequences of postulated accidents will be maintained following the implementation of the proposed EPU. The NRC staff further concludes that the licensee has demonstrated that sufficient cooling exists to ensure the system's design requirements will continue to be met following the implementation of the proposed EPU. Based on this, the NRC staff concludes that the fuel system and associated analyses will continue to meet the requirements of GDCs 4, 23, 25, 26, 27, 28, and 29 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the functional design of the CRDS.

2.8.4.2 Overpressure Protection During Power Operation

Regulatory Evaluation

Overpressure protection for the RCPB during power operation is provided by relief and safety valves and the reactor protection system. The NRC staff's review covers pressurizer relief and safety valves and the piping from these valves to the quench tank and RCS relief and safety valves. The NRC's acceptance criteria are based on (1) GDC-15 for the RCS and associated auxiliary, control, and protection systems being designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including anticipated operational occurrences and (2) GDC-31 for the RCPB being designed with sufficient margin to assure that it behaves in a nonbrittle manner and that the probability of rapidly propagating fracture is minimized. Specific review criteria are contained in SRP Section 5.2.2 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed EPU on the overpressure protection capability of the plant during power operation. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed EPU on pressurization events and overpressure protection features and demonstrated that the plant will continue to have sufficient pressure relief capacity to ensure that pressure limits are not exceeded. Based on this, the NRC staff concludes that the overpressure protection features will continue to provide adequate protection to meet GDC-15 and GDC-31 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to overpressure protection during power operation.

2.8.4.3 Overpressure Protection During Low Temperature Operation

Regulatory Evaluation

Overpressure protection for the reactor coolant pressure boundary (RCPB) during low temperature operation of the plant is provided by pressure-relieving systems that function during the low temperature operation. The NRC staff's review covers relief valves with piping to the quench tank, the makeup and letdown system, and the residual heat removal (RHR) system which may be operating when the primary system is water solid. The NRC's acceptance criteria are based on (1) GDC-15 for the RCS and associated auxiliary, control, and protection systems being designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including anticipated operational occurrences; and (2) GDC-31 for the RCPB being designed with sufficient margin to assure that it behaves in a nonbrittle manner and the probability of rapidly propagating fracture is minimized. Specific review criteria are contained in SRP Section 5.2.2.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed EPU on the overpressure protection capability of the plant during low temperature operation. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed EPU on pressurization events and overpressure protection features and has demonstrated that the plant will continue to have sufficient pressure relief capacity to ensure that pressure limits are not exceeded. Based on this, the NRC staff concludes that the low temperature overpressure protection features will continue to provide adequate protection to meet GDC-15 and GDC-31 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to overpressure protection during low temperature operation.

2.8.4.4 Residual Heat Removal System

Regulatory Evaluation

The RHR system is used to cool down the RCS following shutdown. The RHR system is typically a low pressure system which takes over the shutdown cooling function when the RCS temperature is reduced. The NRC staff's review covers the effect of the proposed EPU on the functional capability of the RHR system to cool the RCS following shutdown and provide decay heat removal. The NRC's acceptance criteria are based on (1) GDC-4 for dynamic effects associated with flow instabilities and loads (e.g., water hammer); (2) GDC-5 which requires that any sharing among nuclear power units of structures, systems, and components important to safety will not significantly impair their safety function; (3) GDC-19 for control room requirements for normal operations and shutdown; and (4) GDC-34 which specifies requirements for an RHR system. Specific review criteria are contained in SRP Section 5.4.7 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed EPU on the RHR system. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed EPU on the system and demonstrated that the RHR system will maintain its ability to cool the RCS following shutdown and provide decay heat removal. Based on this, the NRC staff concludes that the RHR system will continue to meet the requirements of GDCs 4, 5, 19, and 34 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the RHR system.

2.8.5 Accident and Transient Analyses

2.8.5.1. Increase in Heat Removal by the Secondary System

2.8.5.1.1. Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve

Regulatory Evaluation

Excessive heat removal causes a decrease in moderator temperature which increases core reactivity and can lead to a power level increase and a decrease in shutdown margin. Any unplanned power level increase may result in fuel damage or excessive reactor system pressure. Reactor protection and safety systems are actuated to mitigate the transient. The NRC staff's review covers (1) postulated initial core and reactor conditions, (2) methods of thermal and hydraulic analyses, (3) the sequence of events, (4) assumed reactions of reactor systems components, (5) functional and operational characteristics of the reactor protection system, (6) required operator actions, and (7) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC-10 for the RCS being designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operations, including anticipated operational occurrences; (2) GDC-15 for the RCS and its associated auxiliaries being designed with appropriate margin to ensure that the RCPB will not be breached during normal operations, including anticipated operational occurrences; (3) GDC-20 for the reactor protection system being designed to automatically initiate the operation of appropriate systems, including the reactivity control systems, to ensure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including anticipated operational occurrences; and (4) GDC-26 for the reliable control of reactivity changes to ensure that specified acceptable fuel design limits are not exceeded, including anticipated operational occurrences. Specific review criteria are contained in SRP Section 15.1.1-4 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's analyses of the excess heat removal events described above and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the specified acceptable fuel design limits and the RCPB pressure limits will not be exceeded as a result of these events. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 15, 20, and 26 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the events stated.

2.8.5.1.2. Steam System Piping Failures Inside and Outside Containment

Regulatory Evaluation

The steam release resulting from a rupture of a main steam pipe will result in an increase in steam flow, a reduction of coolant temperature and pressure, and an increase in core reactivity. The core reactivity increase may cause a power level increase and a decrease in shutdown margin. Reactor protection and safety systems are actuated to mitigate the transient. The NRC staff's review covers (1) postulated initial core and reactor conditions; (2) methods of thermal and hydraulic analyses; (3) the sequence of events; (4) assumed responses of the reactor coolant and auxiliary systems; (5) functional and operational characteristics of the reactor protection system; (6) required operator actions; (7) core power excursion due to power demand created by excessive steam flow; (8) variables influencing neutronics; and (9) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC-27 and GDC-28 for the RCS being designed with appropriate margin to ensure that the capability to cool the core is maintained; (2) GDC-31 for the RCS being designed with sufficient margin to ensure that the RCPB behaves in a nonbrittle manner and that the probability of a propagating fracture is minimized; and (3) GDC-35 for the reactor cooling system and associated auxiliaries being designed to provide abundant emergency core cooling. Specific review criteria are contained in SRP Section 15.1.5 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's analyses of steam system piping failure events and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the ability to insert control rods is maintained, the RCPB pressure limits will not be exceeded, the RCPB will behave in a nonbrittle manner, the probability of a propagating fracture of the RCPB is minimized, and abundant core cooling will be provided. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 27, 28, 31, and 35 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to steam system piping failures.

2.8.5.2 Decrease in Heat Removal By the Secondary System

2.8.5.2.1. Loss of External Load, Turbine Trip, Loss of Condenser Vacuum, and Steam Pressure Regulatory Failure

Regulatory Evaluation

A number of initiating events may result in unplanned decreases in heat removal by the secondary system. These events result in a sudden reduction in steam flow and consequently result in pressurization events. Reactor protection and safety systems are actuated to mitigate the transient. The NRC staff's review covers the sequence of events, the analytical models used for analyses, the values of parameters used in the analytical models, and the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC-10 for the RCS being designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operations, including anticipated operational occurrences; (2) GDC-15 for the RCS and its associated auxiliaries being designed with appropriate margin to ensure that the RCPB will not be breached during normal operations, including anticipated operational occurrences; and (3) GDC-26 for the reliable control of reactivity changes to ensure that specified acceptable fuel design limits are not exceeded during normal operation, including anticipated operational occurrences. Specific review criteria are contained in SRP Section 15.2.1-5 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's analyses of the decrease in heat removal events described above and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the specified acceptable fuel design limits and the RCPB pressure limits will not be exceeded as a result of these events. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 15, and 26 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the events stated.

2.8.5.2.2 Loss of Nonemergency AC Power to the Station Auxiliaries

Regulatory Evaluation

The loss of nonemergency ac power is assumed to result in the loss of all power to the station auxiliaries and the simultaneous tripping of all reactor coolant circulation pumps. This causes a flow coastdown, as well as a decrease in heat removal by the secondary system, a turbine trip, an increase in pressure and temperature of the coolant, and a reactor trip. Reactor protection and safety systems are actuated to mitigate the transient. The NRC staff's review covers (1) the sequence of events, (2) the analytical model used for analyses, (3) the values of parameters used in the analytical model, and (4) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC-10 for the RCS being designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operation, including anticipated operational occurrences; (2) GDC-15 for the RCS and its associated auxiliaries being designed with appropriate margin to ensure that the RCPB will not be breached during normal operation, including anticipated operational occurrences; and (3) GDC-26 for the reliable control of reactivity changes to assure ensure that specified acceptable fuel design limits are not exceeded during normal operation, including anticipated operational occurrences. Specific review criteria are contained in SRP Section 15.2.6 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's analyses of the loss of nonemergency ac power to station auxiliaries event and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the specified acceptable fuel design limits and the RCPB pressure limits will not be exceeded as a result of this event. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 15, and 26 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the loss of nonemergency ac power to station auxiliaries event.

2.8.5.2.3 Loss of Normal Feedwater Flow

Regulatory Evaluation

A loss of normal feedwater flow could occur from pump failures, valve malfunctions, or a loss of offsite power (LOOP). Loss of feedwater flow results in an increase in reactor coolant temperature and pressure, which eventually requires a reactor trip to prevent fuel damage. Decay heat must be transferred from fuel following a loss of normal feedwater flow. Reactor protection and safety systems are actuated to provide this function and mitigate other aspects of the transient. The NRC staff's review covers (1) the sequence of events, (2) the analytical model used for analyses, (3) the values of parameters used in the analytical model, and (4) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC-10 for the RCS being designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operations, including anticipated operational occurrences; (2) GDC-15 for the RCS and its associated auxiliaries being designed with appropriate margin to ensure that the RCPB will not be breached during normal operations, including anticipated operational occurrences; and (3) GDC-26 for the reliable control of reactivity changes to ensure that specified acceptable fuel design limits are not exceeded during normal operation, including anticipated operational occurrences. Specific review criteria are contained in SRP Section 15.2.7 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's analyses of the loss of normal feedwater flow event and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the specified acceptable fuel design limits and the RCPB pressure limits will not be exceeded as a result of the loss of normal feedwater flow. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 15, and 26 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the loss of normal feedwater flow event.

2.8.5.2.4 Feedwater System Pipe Breaks Inside and Outside Containment

Regulatory Evaluation

Depending upon the size and location of the break and the plant operating conditions at the time of the break, the break could cause either a RCS cooldown (by excessive energy discharge through the break or a RCS heatup (by reducing feedwater flow to the affected RCS). In either case, reactor protection and safety systems are actuated to mitigate the transient. The NRC staff's review covers (1) postulated initial core and reactor conditions, (2) the methods of thermal and hydraulic analyses, (3) the sequence of events, (4) the assumed response of the reactor coolant and auxiliary systems, (5) the functional and operational characteristics of the reactor protection system, (6) required operator actions, and (7) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC-27 and GDC-28 for the RCS being designed with appropriate margin to ensure that the capability to cool the core is maintained; (2) GDC-31 for the RCS being designed with sufficient margin to ensure that the RCPB behaves in a nonbrittle manner and that the probability of a propagating fracture is minimized; and (3) GDC-35 for the reactor cooling system and associated auxiliaries being designed to provide abundant emergency core cooling. Specific review criteria are contained in SRP Section 15.2.8 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's analyses of feedwater system pipe breaks and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the ability to insert control rods is maintained, the RCPB pressure limits will not be exceeded, the RCPB will behave in a nonbrittle manner, the probability of propagating fracture of the RCPB is minimized, and abundant core cooling will be provided. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 27, 28, 31, and 35 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to feedwater system pipe breaks.

2.8.5.3 Decrease in Reactor Coolant System Flow

2.8.5.3.1. Loss of Forced Reactor Coolant Flow

Regulatory Evaluation

A decrease in reactor coolant flow occurring while the plant is at power could result in a degradation of core heat transfer. An increase in fuel temperature and accompanying fuel damage could then result if specified acceptable fuel design limits are exceeded during the transient. Reactor protection and safety systems are actuated to mitigate the transient. The NRC staff's review covers (1) the postulated initial core and reactor conditions, (2) the methods of thermal and hydraulic analyses, (3) the sequence of events, (4) assumed reactions of reactor systems components, (5) the functional and operational characteristics of the reactor protection system, (6) required operator actions, and (7) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC-10 for the RCS being designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operations, including anticipated operational occurrences; (2) GDC-15 for the RCS and its associated auxiliaries being designed with appropriate margin to ensure that the RCPB will not be breached during normal operations, including anticipated operational occurrences; and (3) GDC-26 for the reliable control of reactivity changes to ensure that specified acceptable fuel design limits are not exceeded during normal operation, including anticipated operational occurrences. Specific review criteria are contained in SRP Section 15.3.1-2 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's analyses of the decrease in reactor coolant flow event and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the specified acceptable fuel design limits and the RCPB pressure limits will not be exceeded as a result of this event. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 15, and 26 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the decrease in reactor coolant flow event.

2.8.5.3.2 Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break

Regulatory Evaluation

The events postulated are an instantaneous seizure of the rotor or break of the shaft of a reactor coolant pump. Flow through the affected loop is rapidly reduced, leading to a reactor and turbine trip. The sudden decrease in core coolant flow while the reactor is at power results in a degradation of core heat transfer, which could result in fuel damage. The initial rate of reduction of coolant flow is greater for the rotor seizure event. However, the shaft break event permits a greater reverse flow through the affected loop later during the transient and, therefore, results in a lower core flow rate at that time. In either case, reactor protection and safety systems are actuated to mitigate the transient. The NRC staff's review covers (1) the postulated initial and long-term core and reactor conditions, (2) the methods of thermal and hydraulic analyses, (3) the sequence of events, (4) the assumed reactions of reactor systems components, (5) the functional and operational characteristics of the reactor protection system, (6) required operator actions, and (7) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC-27 and GDC-28 for the RCS being designed with appropriate margin to ensure that the capability to cool the core is maintained; and (2) GDC-31 for the RCS being designed with sufficient margin to ensure that the RCPB behaves in a nonbrittle manner and that the probability of propagating fracture is minimized. Specific review criteria are contained in SRP Section 15.3.3-4 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's analyses of the sudden decrease in core coolant flow events and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the ability to insert control rods is maintained, the RCPB pressure limits will not be exceeded, the RCPB will behave in a nonbrittle manner, the probability of propagating fracture of the RCPB is minimized, and adequate core cooling will be provided. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 27, 28, and 31 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the sudden decrease in core coolant flow events.

2.8.5.4 Reactivity and Power Distribution Anomalies

2.8.5.4.1. Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition

Regulatory Evaluation

An uncontrolled control rod assembly withdrawal from subcritical or low power startup condition may be caused by a malfunction of the reactor control or rod control systems. This withdrawal will uncontrollably add positive reactivity to the reactor core, resulting in a power excursion. The NRC staff's review covers (1) the description of the causes of the transient and the transient itself, (2) the initial conditions, (3) the reactor parameters used in the analysis, (4) the analytical methods and computer codes used, and (5) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC-10 for the RCS being designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operations, including anticipated operational occurrences; (2) GDC-20 for the reactor protection system being designed to initiate automatically the operation of appropriate systems, including the reactivity control systems, to ensure that specified acceptable fuel design limits are not exceeded during normal operation, including anticipated operational occurrences; and (3) GDC-25 for the functional design of redundant reactivity control systems to assure that specified acceptable fuel design limits are not exceeded in the event of a single malfunction of the reactivity control systems. Specific review criteria are contained in SRP Section 15.4.1 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's analyses of the uncontrolled control rod assembly withdrawal from a subcritical or low power startup condition and concludes that the licensee's analyses have adequately accounted for the changes in core design required for operation of the plant at the proposed power level. The NRC staff also concludes that the licensee's analyses were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure the specified acceptable fuel design limits are not exceeded. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 20, and 25 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the uncontrolled control rod assembly withdrawal from a subcritical or low power startup condition.

2.8.5.4.2 Uncontrolled Control Rod Assembly Withdrawal at Power

Regulatory Evaluation

An uncontrolled control rod assembly withdrawal at power may be caused by a malfunction of the reactor control or rod control systems. This withdrawal will uncontrollably add positive reactivity to the reactor core, resulting in a power excursion. The NRC staff's review covers (1) the description of the causes of the anticipated operational occurrence and the description of the event itself, (2) the initial conditions, (3) the reactor parameters used in the analysis, (4) the analytical methods and computer codes used, and (5) the results of the associated analyses. The NRC's acceptance criteria are based on (1) GDC-10 for the RCS being designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operations, including anticipated operational occurrences; (2) GDC-20 as it relates the reactor protection system being designed to initiate automatically the operation of appropriate systems, including the reactivity control systems, to ensure that specified acceptable fuel design limits are not exceeded during normal operation, including anticipated operational occurrences; and (3) GDC-25 for the functional design of redundant reactivity systems to assure that specified acceptable fuel design limits are not exceeded in the event of a single malfunction of the reactivity control systems. Specific review criteria are contained in SRP Section 15.4.2 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's analyses of the uncontrolled control rod assembly withdrawal at power event and concludes that the licensee's analyses have adequately accounted for the changes in core design required for operation of the plant at the proposed power level. The NRC staff also concludes that the licensee's analyses were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure the specified acceptable fuel design limits are not exceeded. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 20, and 25 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the uncontrolled control rod assembly withdrawal at power event.

2.8.5.4.3 Control Rod Misoperation

Regulatory Evaluation

The NRC staff's review covers the types of control rod misoperations that are assumed to occur, including those caused by a system malfunction or operator error. The review covers (1) descriptions of rod position, flux, pressure, and temperature indication systems, and those actions initiated by these systems (e.g., turbine runback, rod withdrawal prohibit, rod block) which can mitigate the effects or prevent the occurrence of various misoperations; (2) the sequence of events; (3) the analytical model used for analyses; (4) important inputs to the calculations; and (5) the results of the analyses. The NRC's acceptance criteria are based on (1) GDC-10 for the RCS being designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operations, including anticipated operational occurrences; (2) GDC-20 for the reactor protection system being designed to automatically initiate appropriate systems to ensure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences; and (3) GDC-25 for the functional design of redundant reactivity systems to assure that specified acceptable fuel design limits are not exceeded in the event of a single malfunction of the reactivity control systems. Specific review criteria are contained in SRP Section 15.4.3 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's analyses of control rod misoperation events and concludes that the licensee's analyses have adequately accounted for the changes in core design required for operation of the plant at the proposed power level. The NRC staff also concludes that the licensee's analyses were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure the specified acceptable fuel design limits are not exceeded. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 20, and 25 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to control rod misoperation events.

2.8.5.4.4 Startup of an Inactive Loop at an Incorrect Temperature

Regulatory Evaluation

A startup of an inactive loop transient may result in either an increased core flow or the introduction of cooler or deborated water into the core. This event causes an increase in core reactivity due to decreased moderator temperature or moderator boron concentration. The NRC staff's review covers (1) the sequence of events, (2) the analytical model, (3) the values of parameters used in the analytical model, and (4) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC-10 and GDC-20 for the RCS being designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operations, including anticipated operational occurrences; (2) GDC-15 and GDC-28 for the RCS and its associated auxiliaries being designed with appropriate margin to ensure that the RCPB will not be breached during normal operations, including anticipated operational occurrences; and (3) GDC-26 for the reliable control of reactivity changes to ensure that specified acceptable fuel design limits are not exceeded during normal operations, including anticipated operational occurrences. Specific review criteria are contained in SRP Section 15.4.4-5 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's analyses of the inactive loop startup event and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the specified acceptable fuel design limits and the RCPB pressure limits will not be exceeded as a result of this event. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 15, 20, 26, and 28 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the increase in core flow event.

2.8.5.4.5 Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant

Regulatory Evaluation

Unborated water can be added to the RCS, via the chemical and volume control system (CVCS). This may happen inadvertently because of operator error or CVCS malfunction, and cause an unwanted increase in reactivity and a decrease in shutdown margin. The operator must stop this unplanned dilution before the shutdown margin is eliminated. The NRC staff's review covers (1) conditions at the time of the unplanned dilution, (2) causes, (3) initiating events, (4) the sequence of events, (5) the analytical model used for analyses, (6) the values of parameters used in the analytical model, and (7) results of the analyses. The NRC's acceptance criteria are based on (1) GDC-10 for the reactor core and associated coolant, control, and protection systems being designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including anticipated operational occurrences; (2) GDC-15 for the RCS and associated auxiliary, control, and protection systems being designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including anticipated operational occurrences; and (3) GDC-26 for the control rods being capable of reliably controlling reactivity changes to assure that specified acceptable fuel design limits are not exceeded during normal operation, including anticipated operational occurrences. Specific review criteria are contained in SRP Section 15.4.6 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's analyses of the decrease in boron concentration in the reactor coolant due to a CVCS malfunction and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the specified acceptable fuel design limits and the RCPB pressure limits will not be exceeded as a result of this event. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 15, and 26 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the decrease in boron concentration in the reactor coolant due to a CVCS malfunction.

2.8.5.4.6 Spectrum of Rod Ejection Accidents

Regulatory Evaluation

Control rod ejection accidents cause a rapid positive reactivity insertion together with an adverse core power distribution, which could lead to localized fuel rod damage. The NRC staff evaluates the consequences of a control rod ejection accident to determine the potential damage caused to the RCPB and to determine whether the fuel damage resulting from such an accident could impair cooling water flow. The NRC staff's review covers initial conditions, rod patterns and worths, scram worth as a function of time, reactivity coefficients, the analytical model used for analyses, core parameters which affect the peak reactor pressure or the probability of fuel rod failure, and the results of the transient analyses. The NRC's acceptance criteria are based on GDC-28 for ensuring that the effects of postulated reactivity accidents do not result in damage to the RCPB greater than limited local yielding and do not cause sufficient damage to significantly impair the capability to cool the core. Specific review criteria are contained in SRP Section 15.4.8 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's analyses of the rod ejection accident and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that appropriate reactor protection and safety systems will prevent postulated reactivity accidents that could (1) result in damage to the RCPB greater than limited local yielding, or (2) cause sufficient damage that would significantly impair the capability to cool the core. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDC-28 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the rod ejection accident.

2.8.5.5 Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory

Regulatory Evaluation

Equipment malfunctions, operator errors, and abnormal occurrences could cause unplanned increases in reactor coolant inventory. Depending on the boron concentration and temperature of the injected water and the response of the automatic control systems, a power level increase may result and, without adequate controls, could lead to fuel damage or overpressurization of the RCS. Alternatively, a power level decrease and depressurization may result. Reactor protection and safety systems are actuated to mitigate these events. The NRC staff's review covers (1) the sequence of events, (2) the analytical model used for analyses, (3) the values of parameters used in the analytical model, and (4) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC-10 for the RCS being designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operations, including anticipated operational occurrences; (2) GDC-15 for the RCS and its associated auxiliaries being designed with appropriate margin to ensure that the RCPB will not be breached during normal operations, including anticipated operational occurrences; and (3) GDC-26 for the reliable control of reactivity changes to ensure that specified acceptable fuel design limits are not exceeded during normal operation, including anticipated operational occurrences. Specific review criteria are contained in SRP Section 15.5.1-2 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's analyses of the inadvertent operation of ECCS and CVCS event and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the specified acceptable fuel design limits and the RCPB pressure limits will not be exceeded as a result of this event. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 15, and 26 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the inadvertent operation of ECCS and CVCS event.

2.8.5.6 Decrease in Reactor Coolant Inventory

2.8.5.6.1 Inadvertent Opening of Pressurizer Pressure Relief Valve

Regulatory Evaluation

The inadvertent opening of a pressure relief valve results in a reactor coolant inventory decrease and a decrease in RCS pressure. A reactor trip normally occurs due to low RCS pressure. The NRC staff's review covers (1) the sequence of events, (2) the analytical model used for analyses, (3) the values of parameters used in the analytical model, and (4) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC-10 for the RCS being designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operations, including anticipated operational occurrences; (2) GDC-15 for the RCS and its associated auxiliaries being designed with appropriate margin to ensure that the RCPB will not be breached during normal operations, including anticipated operational occurrences; and (3) GDC-26 for the reactivity control systems to provide adequate control of reactivity changes to ensure that the acceptable fuel design limits are not exceeded during normal operations and anticipated transients during normal operations, including anticipated operational occurrences. Specific review criteria are contained in SRP Section 15.6.1 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's analyses of the inadvertent opening of a pressurizer pressure relief valve event and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the specified acceptable fuel design limits and the RCPB pressure limits will not be exceeded as a result of this event. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 15, and 26 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the inadvertent opening of a pressurizer pressure relief valve event.

2.8.5.6.2 Steam Generator Tube Rupture

Regulatory Evaluation

A steam generator tube rupture (SGTR) event causes a direct release of radioactive material contained in the primary coolant to the environment through the ruptured steam generator tube and RCS safety or atmospheric relief valves. Reactor protection and engineered safety features are actuated to mitigate the accident and restrict the offsite dose to within the guidelines of 10 CFR Part 100. The NRC staff's review covers (1) postulated initial core and plant conditions, (2) method of thermal and hydraulic analysis, (3) the sequence of events (assuming with and without offsite power available), (4) assumed reactions of reactor system components, (5) functional and operational characteristics of the reactor protection system, (6) required operator actions consistent with the plant's emergency operating procedures (EOPs), and (7) the results of the accident analysis. A single failure of a mitigating system is assumed for this event. The NRC staff's review for SGTR discussed in this section is focused on the thermal and hydraulic analysis for the SGTR in order to (1) support the review related to 10 CFR Part 100 for radiological consequences, which is discussed in Section 2.7 of this safety evaluation and (2) confirm that RCSs do not experience an overfill. Preventing a RCS overfill is required in order to prevent failure of the main steamlines. Specific review criteria are contained in SRP Section 15.6.3 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's analysis of the SGTR accident and concludes that the licensee's analysis has adequately accounted for operation of the plant at the proposed power level and was performed using acceptable analytical methods and approved computer codes. The NRC staff further concludes that the assumptions used in this analysis are conservative and that the event does not result in an overfill of the RCS. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the SGTR event.

2.8.5.6.3 Emergency Core Cooling System and Loss-of-Coolant Accidents

Regulatory Evaluation

Loss-of-coolant accidents (LOCAs) are postulated accidents that would result from the loss of reactor coolant from piping breaks in the RCPB at a rate in excess of the capability of the normal reactor coolant makeup system. Loss of significant quantities of reactor coolant would prevent heat removal from the reactor core, unless the water is replenished. The reactor protection and ECCS systems are provided to mitigate this accidents. The NRC staff's review covers (1) the licensee's determination of break locations and break sizes; (2) postulated initial conditions; (3) the sequence of events; (4) the analytical model used for analyses, calculations of the reactor power, pressure, flow, and temperature transients; (5) calculations for peak cladding temperature, total oxidation of the cladding, total hydrogen generation, changes in core geometry, and long-term cooling; (6) functional and operational characteristics of the reactor protective and ECCS systems; and (7) required operator actions. The NRC's acceptance criteria are based on (1) 10 CFR 50.46 and Appendix K to 10 CFR Part 50 for the use of an acceptable evaluation model for LOCA analyses and ECCS equipment being provided that refills the vessel in a timely manner for a LOCA; (2) GDC-4 for the dynamic effects associated with flow instabilities and loads (e.g., water hammer); (3) GDC-27 for the ECCS design having the capability to assure that under postulated accident conditions and with appropriate margin for stuck rods, the capability to cool the core is maintained; and (4) GDC-35 for the ECCS being designed to provide an abundance of core cooling to transfer heat from the core at a rate so that fuel and clad damage will not interfere with continued effective core cooling. Specific review criteria are contained in SRP Sections 6.3 and 15.6.5 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's analyses of the LOCA events and the ECCS. The NRC staff concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and that the analyses were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection system and the ECCS will continue to ensure that the peak cladding temperature, total oxidation of the cladding, total hydrogen generation, and changes in core geometry, and long-term cooling will remain within acceptable limits. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 4, 27, 35, and 10 CFR 50.46 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the LOCA.

2.8.5.7 Anticipated Transients Without Scrams

Regulatory Evaluation

Anticipated transients without scram (ATWS) is defined as an anticipated operational occurrence followed by the failure of the reactor portion of the protection system specified in GDC-20. The regulation at 10 CFR 50.62 requires that:

- each PWR must have equipment that is diverse from the reactor trip system to automatically initiate the auxiliary (or emergency) feedwater system and initiate a turbine trip under conditions indicative of an ATWS. This equipment must perform its function in a reliable manner and be independent from the existing reactor trip system, and

- each PWR manufactured by Combustion Engineering (CE) or Babcock and Wilcox (B&W) must have a diverse scram system (DSS). This scram system must be designed to perform its function in a reliable manner and be independent from the existing reactor trip system.

The NRC staff's review is conducted to ensure that the above requirements are satisfied and that the setpoints for the ATWS mitigating system actuation circuitry (AMSAC) and DSS remain valid for the proposed EPU. In addition, for plants where a DSS is not specifically required by 10 CFR 50.62, the NRC staff verifies that the consequences of an ATWS are acceptable. The acceptance criteria is that the peak primary system pressure should not exceed the ASME Service Level C limit of 3200 psig. The peak ATWS pressure is primarily a function of the moderator temperature coefficient (MTC) and the primary system relief capacity. The NRC staff reviews (1) the limiting event determination, (2) the sequence of events, (3) the analytical model and its applicability, (4) the values of parameters used in the analytical model, and (5) the results of the analyses. If the licensee relies upon generic vendor analyses, the NRC staff reviews the licensee's justification of the applicability of those analyses to the plant under review and the operating conditions for the proposed EPU. Review guidance is provided in Matrix 8 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the information submitted by the licensee related to ATWS and concludes that the licensee has adequately accounted for the effects of the proposed EPU on ATWS. The NRC staff concludes that the licensee has demonstrated that the AMSAC **[and DSS]** will continue to meet the requirements of 10 CFR 50.62 following implementation of the proposed EPU. **[*For plants not required to install DSS, use the following sentence:* The licensee has shown that the plant is not required by 10 CFR 50.62 to have a DSS. Additionally, the licensee has demonstrated, through acceptable analyses, that the peak primary system pressure following an ATWS event will remain below the acceptance limit of 3200 psig.]** Based on this, the NRC staff concludes that the plant design will continue to meet the requirements of 10 CFR 50.62 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to ATWS.

2.8.6 Fuel Storage

2.8.6.1 New Fuel Storage

Regulatory Evaluation

Nuclear reactor plants include facilities for the storage of new fuel. The quantity of new fuel to be stored varies from plant to plant, depending upon the specific design of the plant and the individual refueling requirements. The NRC staff's review covers the ability of the storage facilities to maintain the new fuel in a subcritical array during all credible storage conditions. The review focuses on the effect of changes in fuel design on the analyses for the new fuel storage facilities. The NRC's acceptance criteria are based on GDC-62 for the prevention of criticality by physical systems or processes utilizing geometrically safe configurations. Specific review criteria are contained in SRP Section 9.1.1.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's analyses related to the effect of the new fuel on the analyses for the new fuel storage facilities and concludes that the new fuel storage facilities will continue to meet the requirements of GDC-62 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the new fuel storage.

2.8.6.2 Spent Fuel Storage

Regulatory Evaluation

Nuclear reactor plants include storage facilities for the wet storage of spent fuel assemblies. The safety function of the spent fuel pool and storage racks is to maintain the spent fuel assemblies in a safe and subcritical array during all credible storage conditions and to provide a safe means of loading the assemblies into shipping casks. The NRC staff's review covers the effect of the proposed EPU on the criticality analysis (e.g., reactivity of the spent fuel storage array and boraflex degradation or neutron poison efficacy). The NRC's acceptance criteria are based on (1) GDC-4 for the facility being capable to withstand the effects of environmental conditions such that safety functions will not be precluded and (2) GDC-62 for the prevention of criticality by physical systems or processes utilizing geometrically safe configurations. Specific review criteria are contained in SRP Section 9.1.2.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed EPU on the spent fuel storage capability and concludes that the licensee has adequately accounted for the effects of the proposed EPU on the spent fuel rack temperature and criticality analyses. The NRC staff concludes that the spent fuel pool design will continue to ensure an acceptably low temperature and an acceptable degree of subcriticality following implementation of the proposed EPU. Based on this, the NRC staff concludes that the spent fuel storage facilities will continue to meet the requirements of GDCs 4 and 62 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to spent fuel storage.

[2.8.7 Additional Review Areas (Reactor Systems)]

[Insert Regulatory Evaluation, Technical Evaluation, and Conclusion sections as necessary]

INSERT 9

FOR

SECTION 3.3 - PWR TEMPLATE SAFETY EVALUATION

2.9 Source Terms and Radiological Consequences Analyses

2.9.1 Source Terms for Radwaste Systems Analyses

Regulatory Evaluation

The NRC staff reviews the radioactive source term associated with EPUs to ensure the adequacy of the sources of radioactivity used by the licensee as input to verify that the radioactive waste management systems have adequate capacity for the treatment of radioactive liquid and gaseous wastes. The NRC staff's review includes the parameters used to determine (1) the concentration of each radionuclide in the reactor coolant, (2) the fraction of fission product activity released to the reactor coolant, (3) concentrations of all nonfission product radionuclides in the reactor coolant, (4) leakage rates and associated fluid activity of all potentially radioactive water and steam systems, and (5) potential sources of radioactive materials in effluents that are not considered in the plant's [**Updated Safety Analysis Report *or* Updated Final Safety Analysis Report**] related to liquid waste management systems and gaseous waste management systems. The NRC's acceptance criteria for source terms are based on (1) 10 CFR Part 20 for radioactivity in liquid and gaseous effluents released to unrestricted areas; (2) 10 CFR Part 50, Appendix I, for the numerical guides for design objectives and limiting conditions for operation to meet the "as low as reasonably achievable" criterion; and (3) GDC-60 for the radioactive waste management systems being able to control the releases of radioactive liquid and gaseous effluents to the environment. Specific review criteria are contained in SRP Section 11.1.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the radioactive source term associated with the proposed EPU and concludes that the proposed parameters and resultant composition and quantity of radionuclides are appropriate for the evaluation of the radioactive waste management systems. The NRC staff further concludes that the proposed radioactive source term meets the requirements of 10 CFR Part 20, 10 CFR Part 50, Appendix I, and GDC-60. Therefore, the NRC staff finds the proposed EPU acceptable with respect to source terms.

NOTE: Use Sections 2.9.2 and 2.9.3 below if the licensee's radiological consequences analyses are based on an alternative source term.

2.9.2. Radiological Consequences Analyses Using Alternative Source Terms

NOTE: There are two cases that may be encountered here: (1) a licensee may be implementing an alternative source term for the first time, or (2) a licensee may have already fully implemented an alternative source term and is revising the previously approved dose analyses that use alternative source term methodologies. The second paragraph for each heading is only needed for a first-time implementation of an alternative source term (either partial or full implementations). Several accidents may have been analyzed - see corresponding SRP sections for further regulatory evaluation text (to be modified), as needed

Regulatory Evaluation

The NRC staff reviews the DBA radiological consequences analyses. The radiological consequences analyses reviewed are the LOCA, fuel handling accident (FHA), control rod ejection accident (REA), MSLB, SGTR, and locked-rotor accident. The NRC staff's review for each accident analysis includes (1) the sequence of events; and (2) models, assumptions, and parameter inputs used by the licensee for the calculation of the total effective dose equivalent. The NRC's acceptance criteria for radiological consequences analyses using an alternate source term are based on (1) GDC-19 for control room habitability and (2) 10 CFR 50.67 for radiological consequences of a postulated accident. Specific review criteria are contained in SRP Section 15.0.1.

NOTE: Use the following paragraph for a first implementation of an alternative source term:

The NRC staff reviews the implementation of alternative source terms. The NRC's acceptance criteria for implementation of an alternative source term are based on (1) 10 CFR 50.49 for qualification of safety-related equipment with regard to integrated radiation dose during normal and accident conditions; (2) 10 CFR 50.67 for the implementation of an alternative source term in current operating nuclear power plants; (3) GDC-19 for maintaining the control room in a safe, habitable condition under accident conditions by providing adequate protection against radiation and toxic gases; (4) 10 CFR Part 51 for environmental assessments of radioactive material releases during normal and accident conditions; (5) paragraph IV.E.8 of 10 CFR Part 50, Appendix E, for maintaining emergency facilities in a safe, habitable condition under accident conditions by providing adequate protection against radiation and toxic gases; and (6) plant-specific licensing commitments made in response to NUREG-0737 (Items II.B.2, II.B.3, II.F.1, III.D.1.1, III.A.1.2, and III.D.3.4). Specific review criteria are contained in SRP Sections 15.0.1.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has evaluated the licensee's revised accident analyses performed in support of the proposed EPU and concludes that the licensee has adequately accounted for the effects of the proposed EPU. The NRC staff further concludes that the plant site and the dose-mitigating engineered safety features (ESFs) remain acceptable with respect to the radiological consequences of postulated DBAs since the calculated total effective dose equivalent (TEDE) at the exclusion area boundary (EAB), at the low population zone (LPZ) outer boundary, and in the control room meet the exposure guideline values specified in 10 CFR 50.67 and GDC-19, as well as applicable acceptance criteria denoted in SRP 15.0.1. Therefore, the NRC staff finds the licensee's proposed EPU acceptable with respect to the radiological consequences of DBAs.

NOTE: Use the following paragraph for a first implementation of an alternative source term:

The NRC staff has reviewed the alternative source term methodology used by the licensee in evaluating the effects of the proposed EPU and concludes that changes continue to provide a sufficient margin of safety with adequate defense-in-depth to address unanticipated events and to compensate for uncertainties in accident progression, analysis assumptions, and parameter inputs. Therefore, the NRC staff finds the licensee's proposed EPU acceptable with respect to the implementation of an alternative source term.

[2.9.3 Additional Review Areas (Radiological Consequences Analyses)]

[Insert Regulatory Evaluation, Technical Evaluation, and Conclusion sections as necessary]

NOTE: Use Sections 2.9.2 - 2.9.10 below if the licensee's radiological consequences analyses are not based on an alternative source term (i.e., if the analyses are based on traditional source term, based on TID-14844)

2.9.2. <u>Radiological Consequences of Main Steamline Failures Outside Containment</u>

<u>Regulatory Evaluation</u>

The NRC staff reviews the analyses of the radiological consequences of a main steamline break (MSLB) outside the containment. The NRC staff's review includes (1) the sequence of events, models and assumptions used by the licensee for the calculation of the radiological doses; (2) evaluation of the TSs on the primary and secondary coolant iodine activities; and (3) determination of reactor coolant iodine concentration corresponding to a preaccident iodine spike and a concurrent iodine spike. The NRC's acceptance criteria for the radiological consequences of an MSLB outside containment are based on (1) GDC-19 for control room habitability and (2) 10 CFR Part 100 for the radiological consequences of a postulated accident. Specific review criteria are contained in SRP Sections 6.4 and 15.1.5.A, and other guidance provided in Matrix 9 of RS-001.

<u>Technical Evaluation</u>

[Insert technical evaluation.]

<u>Conclusion</u>

The NRC staff has evaluated the licensee's revised accident analyses for the radiological consequences of an MSLB outside containment and concludes that the licensee has adequately accounted for the effects of the proposed EPU on these analyses. The NRC staff further concludes that the plant site and the dose-mitigating ESFs remain acceptable with respect to the radiological consequences of a postulated MSLB outside containment since the calculated whole-body and thyroid doses at the exclusion area boundary (EAB) and the low population zone (LPZ) outer boundary meet the exposure guideline values specified in 10 CFR 100.11 (assuming a preaccident iodine spike) and are a small fraction of the Part 100 values for the concurrent iodine spike. The NRC staff also concludes that the control room meets the dose requirements of GDC-19 for DBAs. Therefore, the NRC staff finds the licensee's proposed EPU acceptable with respect to the radiological consequences of MSLB accidents outside the containment.

2.9.3 Radiological Consequences of a Reactor Coolant Pump Locked-Rotor Accident

Regulatory Evaluation

The NRC staff reviews the analyses of the radiological consequences of a reactor coolant pump locked-rotor accident. The review includes (1) determination of a need for a radiological consequences analysis; and (2) the sequence of events, models and assumptions used by the licensee for the calculation of radiological doses. The NRC's acceptance criteria for the radiological consequences of a reactor coolant pump locked-rotor accident are based on (1) GDC-19 for control room habitability and (2) 10 CFR Part 100 for the radiological consequences of a postulated accident. Specific review criteria are contained in SRP Sections 6.4 and 15.3.3-15.3.4; and other guidance provided in Matrix 9 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has evaluated the licensee's revised analyses for the radiological consequences of a reactor coolant pump locked rotor and concludes that the licensee has adequately accounted for the effects of the proposed EPU on these analyses. The NRC staff further concludes that the plant site and the dose-mitigating ESFs remain acceptable with respect to the radiological consequences of a postulated locked-rotor accident since the calculated whole-body and thyroid doses at the EAB and the LPZ outer boundary are a small fraction of exposure guideline values specified in 10 CFR 100.11. The NRC staff also concludes that the control room meets the dose requirements of GDC-19 for DBAs. Therefore, the NRC staff finds the licensee's proposed EPU acceptable with respect to the radiological consequences of a locked-rotor accident.

2.9.4 Radiological Consequences of a Control Rod Ejection Accident

Regulatory Evaluation

The NRC staff reviews the analyses of the radiological consequences of a control rod ejection accident. The NRC staff's review includes the plant response to a control rod ejection accident and the calculation of radiological doses at the EAB and LPZ outer boundary and in the control room due to the releases resulting from a rod ejection accident. The purpose of the NRC staff's review is to (1) ensure that plant's procedures for recovery from a rod ejection accident and the plant's TSs are properly taken into account in computing the doses and (2) compare the calculated doses against the appropriate guidelines. The NRC's acceptance criteria for the radiological consequences of a control rod ejection accident are based on (1) GDC-19 for control room habitability and (2) 10 CFR Part 100 for mitigating the radiological consequences of an accident. Specific review criteria are contained in SRP Sections 6.4 and 15.4.8.A, and other guidance provided in Matrix 9 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has evaluated the licensee's revised accident analyses for the radiological consequences of a rod ejection accident and concludes that the licensee has adequately accounted for the effects of the proposed EPU on these analyses. The NRC staff further concludes that the plant site and the dose-mitigating ESFs remain acceptable with respect to the radiological consequences of a postulated control rod ejection accident since the calculated whole-body and thyroid doses at the EAB and the LPZ outer boundary are well within the exposure guideline values specified in 10 CFR 100.11. The NRC staff also concludes that the control room meets the dose requirements of GDC-19 for DBAs. Therefore, the NRC staff finds the licensee's proposed EPU acceptable with respect to the radiological consequences of a control rod ejection accident.

2.9.5 Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment

Regulatory Evaluation

The NRC staff reviews the analyses of the radiological consequences of failures outside the containment of small lines connected to the primary coolant pressure boundary (e.g., instrument lines and sample lines). The NRC staff's review includes (1) the identification of small lines postulated to fail and the isolation provisions for these lines; (2) the failure scenario; (3) the models and assumptions for the calculation of the radiological doses for the postulated failure; and (4) an evaluation of the primary coolant iodine activity, including the effects of a concurrent iodine spike, and the TSs for the reactor coolant iodine activity. The NRC's acceptance criteria for the radiological consequences of the failure of small lines carrying primary coolant outside containment are based on (1) GDC-19 for control room habitability and (2) GDC-55 for the isolation requirements of small-diameter lines connected to the primary system that are acceptable on the basis of meeting 10 CFR 100.11. Specific review criteria are contained in SRP Sections 6.4 and 15.6.2, and other guidance provided in Matrix 9 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has evaluated the licensee's revised accident analyses for the radiological consequences of failures outside the containment of small lines connected to the primary coolant pressure boundary and concludes that the licensee has adequately accounted for the effects of the proposed EPU on these analyses. The NRC staff further concludes that the plant site and the dose-mitigating ESFs remain acceptable with respect to the radiological consequences of a postulated failure outside the containment of a small line carrying reactor coolant since the calculated whole-body and thyroid doses at the EAB and the LPZ outer boundary are substantially below the exposure guideline values of 10 CFR 100.11. The NRC staff also concludes that the control room meets the dose requirements of GDC-19 for DBAs. Therefore, the NRC staff finds the licensee's proposed EPU acceptable with respect to the radiological consequences of failures outside the containment of small lines connected to the primary coolant pressure boundary.

2.9.6 Radiological Consequences of Steam Generator Tube Rupture

Regulatory Evaluation

The NRC staff reviews the analysis of the radiological consequences of a postulated steam generator tube rupture (SGTR). The NRC staff's review includes (1) a review of the sequence of events and plant procedures for recovery from the accident to ensure that the most severe case of radioactive releases has been considered; (2) a review of the models and assumptions for the calculation of the radiological doses for the postulated accident; (3) an evaluation of the TSs on the primary and secondary coolant iodine activity concentration; and (4) an evaluation of the radiological consequences of an SGTR concurrent with a loss of offsite power and the most limiting single failure. The NRC staff's review includes two cases for the reactor coolant iodine concentration corresponding to a preaccident iodine spike and a concurrent iodine spike. The NRC's acceptance criteria for the radiological consequences of an SGTR are based on (1) GDC-19 for control room habitability and (2) 10 CFR Part 100 for mitigating the radiological consequences of an accident. Specific review criteria are contained in SRP Sections 6.4 and 15.6.3, and other guidance provided in Matrix 9 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has evaluated the licensee's revised accident analyses for the radiological consequences of an SGTR and concludes that the licensee has adequately accounted for the effects of the proposed EPU on these analyses. The NRC staff further concludes that the plant site and the dose-mitigating ESFs remain acceptable with respect to the radiological consequences of an SGTR accident since the calculated whole-body and thyroid doses at the EAB and the LPZ outer boundary do not exceed the exposure guideline values of 10 CFR 100.11 (assuming a preaccident iodine spike) and are a small fraction of the Part 100 values for the concurrent iodine spike. The NRC staff also concludes that the control room meets the dose requirements of GDC-19 for DBAs. Therefore, the NRC staff finds the licensee's proposed EPU acceptable with respect to the radiological consequences of an SGTR.

2.9.7 Radiological Consequences of a Design-Basis Loss-of-Coolant Accident

Regulatory Evaluation

The NRC staff reviews the analyses of the radiological consequences of a design-basis LOCA. The review includes a summary review of the doses from the hypothetical design-basis LOCA and a specific review of the doses from containment leakage and leakage from ESF components outside containment that contribute to the total LOCA doses. The NRC staff's review also includes (1) the methodology and results of calculations of the radiological consequences resulting from containment and ESF component leakage following a hypothetical LOCA; and (2) an assessment of the containment with respect to the assumptions and the input parameters for the dose calculations. The NRC staff's calculations are based on pertinent information in the [**Updated Safety Analysis Report** ***or*** **Updated Final Safety Analysis Report**] and considers the NRC staff's evaluation of dose-mitigating ESFs. The NRC's acceptance criteria for the radiological consequences of a design-basis LOCA are based on (1) GDC-19 for control room habitability and (2) 10 CFR Part 100 for mitigating the radiological consequences of an accident. Specific review criteria are contained in SRP Section 6.4 and Appendices A and B of SRP Section 15.6.5, and other guidance provided in Matrix 9 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has evaluated the licensee's revised accident analyses for the radiological consequences of a design-basis LOCA and concludes that the licensee has adequately accounted for the effects of the proposed EPU on these analyses. The NRC staff further concludes that the plant site and the dose-mitigating ESFs remain acceptable with respect to the radiological consequences of a design-basis LOCA since the calculated whole-body and thyroid doses at the EAB and the LPZ outer boundary do not exceed the exposure guideline values of 10 CFR 100.11 and the calculated doses in the control room meet the requirements of GDC-19. Therefore, the NRC staff finds the licensee's proposed EPU acceptable with respect to the radiological consequences of a design-basis LOCA.

2.9.8 Radiological Consequences of Fuel Handling Accidents

Regulatory Evaluation

The NRC staff reviews the analyses of the radiological consequences of a postulated FHA. The purpose of this review is to evaluate the adequacy of system design features and plant procedures provided for the mitigation of the radiological consequences of accidents that involve damage to spent fuel. Such accidents include the dropping of a single fuel assembly and handling tool or a heavy object onto other spent fuel assemblies. Such accidents may occur inside the containment, along the fuel transfer canal, and in the fuel building. The NRC staff's review includes (1) the sequence of events, models, and assumptions used by the licensee for the calculation of radiological doses; (2) the adequacy of the ESFs provided for the purpose of mitigating potential accident doses; and (3) the containment ventilation system with respect to its function as a dose-mitigating ESF system, including the radiation detection system on the containment purge/vent lines for those plants that will vent or purge the containment during fuel handling operations. The NRC's acceptance criteria for the radiological consequences of FHAs are based on (1) GDC-19 for control room habitability; (2) GDC-61 for appropriate containment, confinement, and filtering systems; and (3) 10 CFR Part 100 for the calculated radiological consequences of FHAs. Specific review criteria are contained in SRP Sections 6.4 and 15.7.4, and other guidance provided in Matrix 9 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has evaluated the licensee's revised accident analyses for the radiological consequences of FHAs and concludes that the licensee has adequately accounted for the effects of the proposed EPU on these analyses. The NRC staff further concludes that the plant site and the dose-mitigating ESFs remain acceptable with respect to the radiological consequences of a postulated FHA since the calculated whole-body and thyroid doses at the EAB and the LPZ boundary are well within the exposure guideline values of 10 CFR 100.11 and GDC-61. The NRC staff also concludes that the control room meets the dose requirements of GDC-19 for DBAs. Therefore, the NRC staff finds the licensee's proposed EPU acceptable with respect to the radiological consequences of FHAs.

2.9.9 Radiological Consequences of Spent Fuel Cask Drop Accidents

Regulatory Evaluation

The NRC staff reviews the analyses of the radiological consequences of the release of fission products from irradiated fuel in a spent fuel cask that is postulated to drop during cask handling operations. The NRC staff's review is conducted to verify various design and operations aspects of the system. The NRC staff's review includes (1) determining a need for a design-basis radiological analysis; (2) sequence of events, models and assumptions used by the licensee for the calculation of the radiological doses; and (3) comparing the calculated doses to exposure guidelines to determine the acceptability of the EAB and LPZ outer boundary distances and to confirm the adequacy of ESFs provided for the purpose of mitigating potential doses from spent fuel cask drop accidents, including the effects on control room habitablity. The NRC's acceptance criteria for the radiological consequences of spent fuel cask drop accidents are based on (1) GDC-19 for control room habitability; (2) GDC-61 for appropriate containment, confinement and filtering systems; and (3) 10 CFR Part 100 for the calculated radiological consequences of a spent fuel cask drop accident. Specific review criteria are contained in SRP Sections 6.4 and 15.7.5, and other guidance provided in Matrix 9 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has evaluated the licensee's revised accident analyses for the radiological consequences of a spent fuel cask drop accident and concludes that the licensee has adequately accounted for the effects of the proposed EPU on these analyses. The NRC staff further concludes that the plant site and the dose-mitigating ESFs remain acceptable with respect to the radiological consequences of a postulated spent fuel cask drop accident since the calculated whole-body and thyroid doses at the EAB and LPZ outer boundary are well within the exposure guideline values of 10 CFR 100.11 and GDC-61. The NRC staff also concludes that the control room meets the dose requirements of GDC-19 for DBAs. Therefore, the NRC staff finds the licensee's proposed EPU acceptable with respect to the radiological consequences of spent fuel cask drop accidents.

[2.9.10 Additional Review Areas (Source Terms and Radiological Consequences Analyses)]

[Insert Regulatory Evaluation, Technical Evaluation, and Conclusion sections as necessary]

INSERT 10

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SECTION 3.3 - PWR TEMPLATE SAFETY EVALUATION

2.10 Health Physics

2.10.1 Occupational and Public Radiation Doses

Regulatory Evaluation

The NRC staff conducts its review in this area to ascertain what overall effects the proposed EPU will have on both occupational and public radiation doses and to determine that the licensee has taken the necessary steps to ensure that any dose increases will be maintained as low as is reasonably achievable. The NRC staff's review includes an evaluation of any increases in radiation sources and how this may affect plant area dose rates, plant radiation zones, and plant area accessibility. The NRC staff evaluates how personnel doses needed to access plant vital areas following an accident are affected. The NRC staff considers the effects of the proposed EPU on plant effluent levels and any effect this increase may have on radiation doses at the site boundary. The NRC's acceptance criteria for occupational and public radiation doses are based on 10 CFR Part 20 and GDC-19. Specific review criteria are contained in SRP Sections 12.2, 12.3-12.4, and 12.5, and other guidance provided in Matrix 10 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on radiation source terms and plant radiation levels. The NRC staff concludes that the licensee has taken the necessary steps to ensure that any increases in radiation doses will be maintained as low as reasonably achievable. The NRC staff further concludes that the proposed EPU meets the requirements of 10 CFR Part 20 and GDC-19. Therefore, the NRC staff finds the licensee's proposed EPU acceptable with respect to radiation protection and ensuring that occupational radiation exposures will be maintained as low as reasonably achievable.

[2.10.2 Additional Review Areas (Health Physics)]

[Insert Regulatory Evaluation, Technical Evaluation, and Conclusion sections as necessary]

INSERT 11

FOR

SECTION 3.3 - PWR TEMPLATE SAFETY EVALUATION

2.11 Human Performance

2.11.1 Human Factors

Regulatory Evaluation

The area of human factors deals with programs, procedures, training, and plant design features related to operator performance during normal and accident conditions. The NRC staff's human factors evaluation is conducted to ensure that operator performance is not adversely affected as a result of system changes required for the proposed EPU. The NRC staff's review covers changes to operator actions, human-system interfaces, and procedures and training required for the proposed EPU. The NRC's acceptance criteria for human factors are based on GDC-19, 10 CFR 50.54(i) and (m), 10 CFR 50.59, 10 CFR 50.120, 10 CFR Part 55, and GL 82-33. Specific review criteria are contained in SRP Sections 13.2.1, 13.2.2, 13.5.2.1, and 18.0.

Technical Evaluation

The NRC staff has developed a standard set of questions for the review of the human factors area. The licensee has addressed these questions in its application. Following are the NRC staff's questions, the licensee's responses, and the NRC staff's determination of acceptability.

1. Changes in Emergency and Abnormal Operating Procedures

   Describe how the proposed EPU will change the plant emergency and abnormal operating procedures. (SRP Section 13.5.2.1)

[Insert licensee's response followed by additional NRC staff discussion if necessary]

2. Changes to Operator Actions Sensitive to Power Uprate

   Describe any new operator actions required as a result of the proposed EPU. Describe changes to any current operator actions related to emergency or abnormal procedures that will occur as a result of the proposed EPU. (SRP Section 18.0)

   (i.e., Identify and describe operator actions that will require additional response time or will have reduced time available. Your response should address any operator workarounds that might affect these response times. Identify any operator actions that are being automated or being changed from automatic to manual as a result of the power uprate. Provide justification for the acceptability of these changes).

[Insert licensee's response followed by additional NRC staff discussion if necessary]

3. Changes to Control Room Controls, Displays and Alarms

   Describe any changes the proposed EPU will have on the operator interfaces for control room controls, displays, and alarms. For example, what zone markings (e.g. normal, marginal and out-of-tolerance ranges) on meters will change? What setpoints will change?

How will the operators know of the change? Describe any controls, displays, alarms that will be upgraded from analog to digital instruments as a result of the proposed EPU and how operators were tested to determine they could use the instruments reliably. (SRP Section 18.0)

[Insert licensee's response followed by additional NRC staff discussion if necessary]

4. Changes on the Safety Parameter Display System

   Describe any changes the proposed EPU will have on the safety parameter display system. How will the operators know of the changes? (SRP Section 18.0)

[Insert licensee's response followed by additional NRC staff discussion if necessary]

5. Changes to the Operator Training Program and the Control Room Simulator

   Describe any changes the proposed EPU will have on the operator training program and the plant referenced control room simulator, and provide the implementation schedule for making the changes. (SRP Sections 13.2.1 and 13.2.2)

[Insert licensee's response followed by additional NRC staff discussion if necessary]

Conclusion

The NRC staff has reviewed the changes to operator actions, human-system interfaces, procedures and training required for the proposed EPU and concludes that the licensee has (1) appropriately accounted for the effects of the proposed EPU on the available time for operator actions and (2) taken appropriate actions to ensure that operator performance is not adversely affected by the proposed EPU. The NRC staff further concludes that the licensee will continue to meet the requirements of GDC-19, 10 CFR 50.54(i) and (m), 10 CFR 50.59, 10 CFR 50.120, and 10 CFR Part 55 following implementation of the proposed EPU. Therefore, the NRC staff finds the licensee's proposed EPU acceptable with respect to the human factors aspects of the required system changes.

[2.11.2 Additional Review Areas (Human Performance)]

[Insert Regulatory Evaluation, Technical Evaluation, and Conclusion sections as necessary]

INSERT 12

FOR

SECTION 3.3 - PWR TEMPLATE SAFETY EVALUATION

2.12 Power Ascension and Testing Plan

2.12.1 Approach to EPU Power Level and Test Plan

Regulatory Evaluation

The purpose of the EPU test program is to demonstrate that SSCs will perform satisfactorily in service at the proposed EPU power level. The test program also provides additional assurance that the plant will continue to operate in accordance with design criteria at EPU conditions. The NRC staff's review includes an evaluation of: (1) plans for the initial approach to the proposed maximum licensed thermal power level, including verification of adequate plant performance, (2) transient testing necessary to demonstrate that plant equipment will perform satisfactorily at the proposed increased maximum licensed thermal power level, and (3) the test program's conformance with applicable regulations. The NRC's acceptance criteria for the EPU test program are based on 10 CFR Part 50, Appendix B, Criterion XI, as it relates to establishment of a test program to demonstrate that SSCs will perform satisfactorily in service. Specific review criteria are contained in SRP Section 14.2.1.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The staff has reviewed the EPU test program, including plans for the initial approach to the proposed maximum licensed thermal power level, transient testing necessary to demonstrate that plant equipment will perform satisfactorily at the proposed increased maximum licensed thermal power level, and the test program's conformance with applicable regulations. The staff concludes that the proposed EPU test program provides adequate assurance that the plant will operate in accordance with design criteria and that SSCs affected by the EPU or modified to support the proposed power increase will perform satisfactorily in service. Further, the staff finds that there is reasonable assurance that the EPU testing program satisfies the requirements of 10 CFR Part 50, Appendix B, Criterion XI. Therefore, the NRC staff finds the proposed EPU test program acceptable.

[2.12.2 Additional Review Areas (Power Ascension and Testing Plan)]

[Insert Regulatory Evaluation, Technical Evaluation, and Conclusion sections as necessary]

INSERT 13

FOR

SECTION 3.3 - PWR TEMPLATE SAFETY EVALUATION

2.13 Risk Evaluation

2.13.1 Risk Evaluation of Extended Power Uprate

Regulatory Evaluation

A risk evaluation is conducted to (1) demonstrate that the risks associated with the proposed EPU are acceptable and (2) determine if "special circumstances" are created by the proposed EPU. As described in Appendix D of SRP Chapter 19, special circumstances are any issues that would potentially rebut the presumption of adequate protection provided by the licensee to meet the deterministic requirements and regulations. The NRC staff's review covers the impact of the proposed EPU on core damage frequency (CDF) and large early release frequency (LERF) for the plant due to changes in the risks associated with internal events, external events, and shutdown operations. In addition, the NRC staff's review covers the quality of the risk analyses used by the licensee to support the application for the proposed EPU. This includes a review of licensee actions to address issues or weaknesses that may have been raised in previous NRC staff reviews of the licensee's individual plant examinations (IPEs) and individual plant examinations of external events (IPEEE), or by an industry peer review. The NRC's risk acceptability guidelines are contained in RG 1.174. Specific review guidance is contained in Matrix 13 of RS-001 and its attachments.

Technical Evaluation

[Insert technical evaluation]

Conclusion

The NRC staff has reviewed the licensee's assessment of the risk implications associated with the implementation of the proposed EPU and concludes that the licensee has adequately modeled and/or addressed the potential impacts associated with the implementation of the proposed EPU. The NRC staff further concludes that the results of the licensee's risk analysis indicate that the risks associated with the proposed EPU are acceptable and do not create the "special circumstances" described in Appendix D of SRP Chapter 19. Therefore, the NRC staff finds the risk implications of the proposed EPU acceptable.

[2.13.2 Additional Review Areas (Risk Evaluation)]

[Insert Regulatory Evaluation, Technical Evaluation, and Conclusion sections as necessary]

SECTION 4

INSPECTION GUIDANCE

4.1 Inspection Requirements

Inspection Procedure (IP) 71004, "Power Uprates," describes the inspections required for power uprate related activities and provides guidance for the inspectors to use in conducting these inspections. In addition, the "Recommended Areas for Inspection" section of the final safety evaluation approving an EPU should be considered by inspectors when selecting a sample for implementing IP 71004. The recommendations in the final safety evaluation do not constitute inspection requirements, but are provided to give the inspectors insight into important bases the NRC staff used for approving the EPU.

ATTACHMENT 2

Public Comments Received on Draft RS-001

PUBLIC COMMENTS RECEIVED ON DRAFT RS-001

NUMBER	COMMENT	SOURCE & DATE RECEIVED	RESPONSE
1	Use of Precedent A significant body of Extended Power Uprate (EPU) precedent exists. There are several applications and NRC approvals on record. It would be helpful if a "list of precedents" were maintained either in RS-001 or on the NRC Website.	NEI March 31, 2003	We agree. The NRC's power uprate Web site provides a list of license amendments that approved power uprates, along with references to associated correspondence (i.e., applications, supplements). RS-001 was modified to provide a reference to the NRC's power uprate Web site (See the Purpose section of RS-001). Industry service organizations and vendors may also keep such information.
2	Use of NRC-Approved Topical Reports Where an NRC-approved Topical Report is used as the licensing basis for a plant-specific EPU submittal, RS-001 should not be used by the NRC staff as the basis for expanding or re-reviewing the processes, scope, issues, and topics already reviewed and approved during the NRC's Topical Report review and approval process. RS-001 should not be used as the basis for Requests for Additional Information (RAIs) about subjects in a licensee's application that were dispositioned during the NRC staff's approval of EPU-related Topical Reports. The objective of this comment is to preclude RS-001 from inadvertently conflicting with or undermining the long-standing Topical Report review and approval process.	NEI March 31, 2003	We agree with the statement that RS-001 should not undermine the long-standing topical report review and approval process. The NRC staff will use RS-001 for reviewing all EPU applications. For areas where a licensee references approved topical reports in its application, the NRC staff will utilize the approved topical reports in its reviews and will state so in the safety evaluation for the plant under review. RS-001 will not conflict with the topical report process. RS-001 was modified to convey this expectation (See the Purpose section of RS-001).
3	NRC Fee-billing Practices The NEI Licensing Action Task Force (LATF) has initiated a dialogue with the NRC LATF on the subject of NRC fee-billing practices. Specifically, the NEI LATF has requested that NRC consider including the number of review hours charged by Branch and by reviewer for each project with an NRC TAC number. We understand this data is collected by NRC, and including it on the invoices will enable licensees to accurately budget for NRC Part 170 review fees.	NEI March 31, 2003	We understand that this is being handled through the LATF.
4	Backfit Rule (10 CFR 50.109) Given that all plants have plant-specific design features to some extent, the use of RS-001 as a review "standard" may lead to backfit issues. The RS-001 should address this point in some manner. The users of RS-001 need to be mindful of the backfitting constraints articulated by 10 CFR 50.109.	NEI March 31, 2003	We agree. RS-001 encourages licensees to identify differences between their plant's licensing basis and the criteria in the review standard. The NRC staff has and will continue to review plants against theirs licensing bases. Additional clarification related to this comment was added in RS-001 (See the Purpose section of RS-001). (This comment is similar to Comments 7, 10, and 16 below.)

NUMBER	COMMENT	SOURCE & DATE RECEIVED	RESPONSE
5	Management Oversight To supplement the NRR review guidance in LIC-101 (License Amendment Review Procedures, Revision 1), NEI recommends that the role of management in the oversight of NRC staff reviews of EPU applications be summarized and emphasized in Section 1 of RS-001. The regulatory review of an EPU application is an important and resource-intensive activity that warrants additional management emphasis to maximize efficiency and effectiveness.	NEI March 31, 2003	The NRC exercises appropriate management oversight of power uprate reviews. The NRC staff developed an effectiveness and efficiency plan for power uprates and provided this plan to the Commission via SECY-02-0115, "Effectiveness And Efficiency Plan For Power Uprates," dated June 27, 2002. RS-001 is merely one component of the effectiveness and efficiency plan. The use of several status reports has been implemented at the NRC to ensure that appropriate management oversight is provided for power uprate reviews.
6	Sub-section 2.1, Reviewing Extended Power Uprate Applications The review standard suggests that licensees complete several matrices [scope and associated technical review guidance] to identify differences between the Standard Review Plan (NUREG 0800) and the plant's licensing basis. This imposes a burden on licensees to research and prepare the matrices, and could be interpreted to include validation documentation. Licensee preparation could involve significant resources, depending on the level of detail. To avoid the need for excessive documentation, the comparison should be limited to analyses and evaluations submitted for NRC review. Typically these are areas that are not bounded at the current power level or that have a reduction in design margin.	NEI March 31, 2003	RS-001 identifies the areas the NRC staff believes should be addressed in a power uprate application. When a licensee evaluates an area identified in RS-001 and concludes that it is bounded by existing analyses of record, the area and licensee's evaluation of it should still be discussed in sufficient detail to demonstrate to the NRC staff that the licensee's evaluation adequately considered important potential impacts of the power uprate. This will involve identification of the licensing basis against which the evaluation was performed. To achieve efficiency in the NRC staff's review of the application, licensees should complete the matrices for such areas and provide the completed matrices with the application as suggested in RS-001.
7	The matrices contain a column for "other guidance," such as Regulatory Guides, which are not compliance documents unless the applicant has explicitly committed to them and incorporated them into the licensing basis of the plant.	NEI March 31, 2003	RS-001 encourages licensees to identify differences between their plant's licensing basis and the criteria in the review standard. The NRC staff plans to review a plant's power uprate application against the plant's licensing basis. Additional clarification related to this comment was added in RS-001 (See the Purpose section of RS-001). (This comment is similar to Comment 4 above and Comments 10 and 16 below.)
8	Sub-section 2.1, Step 2 – Paragraph (1) and Step 3 – Paragraph (3) These paragraphs encourages licensees "to complete the matrices as part of their application as a quality check to assure that all necessary information has been provided and properly represented, thereby avoiding potential delays, and improving the efficiency of the staff's review." The potential effect of this statement is to establish RS-001 as a de facto "compliance standard" for NRC staff reviewers to use in judging the acceptability of the form and content of an EPU application. Clearly, RS-001 is not a regulatory requirement. It is one alternative for compiling the information needed by the NRC staff to review an EPU application.	NEI March 31, 2003	RS-001 is not a regulatory requirement. However, we believe that significant benefits can be achieved from standardization of applications and reviews. RS-001 provides a mechanism for doing this. RS-001 provides guidance to the NRC staff and licensee on the scope and methods to be used for reviewing EPU applications. RS-001 helps the NRC staff standardize its review and enables licensees to prepare complete applications, both of which could result in a reduction in requests for additional information (RAIs) and an increase in the effectiveness and efficiency of the NRC staff's reviews. Therefore, while RS-001 is not a regulatory requirement, the NRC staff encourages licensees to use it in preparing their EPU applications in order to allow improvements in the overall efficiency of the review of such applications.

NUMBER	COMMENT	SOURCE & DATE RECEIVED	RESPONSE
9	Sub-section 2.1, Step 2 – Paragraph (3) This paragraph states that NRC reviewers should "Use the 'Acceptance Review' column of the matrices as a checklist to document whether the licensee has addressed the areas of review in sufficient detail to allow the staff to proceed with its detailed review" *[emphasis added]*. We see potential problems with the interpretation by individual reviewers of the phrase "in sufficient detail." RS-001 should include additional commentary on what constitutes sufficient detail in the context of an EPU review.	NEI March 31, 2003	Based on experience with acceptance reviews, the NRC staff does not believe that there are any significant problems in this area. Licensees should provide adequate detail such that a reasonable engineer is able to arrive at a similar finding as that made in the licensee's application.
10	Sub-Section 2.1, Selected Matrices Several matrices seem to impose universal acceptance criteria. For example, Matrix 6 (Reactor Systems), Note 8 (inadvertent operation of ECCS), stipulates that non-safety-grade pressure-operated relief valves should not be credited for event mitigation and pressurizer level should not be allowed to reach a pressurizer water-solid condition. The applicability of such a criterion is a function of the licensing-basis analysis and testing that was performed. NEI recommends that NRC management provide the necessary oversight to ensure that acceptance criteria are based on the documented licensing basis.	NEI March 31, 2003	RS-001 encourages licensees to identify differences between their plant's licensing basis and the criteria in the review standard. The NRC staff plans to review a plant's power uprate application against the plant's licensing basis. Additional clarification related to this comment was added in RS-001 (See the Purpose section of RS-001). (This comment is similar to Comments 4 and 7 above and Comment 16 below).

NUMBER	COMMENT	SOURCE & DATE RECEIVED	RESPONSE
11	Sub-section 2.1, Matrix 4 – Instrumentation and Controls RS-001 discusses audits of licensee calculations, and seems to make such audits mandatory rather than optional. For example, Matrix 4 relating to I&C setpoints "requires" an audit of at least one instrument setpoint calculation to check the application of the methodology. NEI recommends that RS-001 stipulate the audits as optional, rather than mandatory. Also, audits should be limited to verifying the proper application of a methodology and should not be used to re-open an NRC-approved methodology for further staff review.	NEI March 31, 2003	The guidance for independent calculations was developed to ensure that it captures the NRC staff's intent for performing independent calculations. The NRC staff's need to perform independent calculations was based on the type of calculations performed and the potential impact of the power uprate on those calculations. As a result, the guidance for independent calculations is topic-specific. For example, the NRC staff (1) identified the specific calculations that it will perform for materials and chemical engineering (see Attachment 1 to Matrix 1 of RS-001), (2) provided guidelines for determining whether independent analyses will be needed for reactor systems (see Attachment 1 to Matrix 8 of RS-001) and radiological dose consequences (see Attachment 1 to Matrix 9 of RS-001), and (3) provided general guidance leaving independent calculations optional for mechanical and civil engineering (see Attachment 1 to Matrix 2 of RS-001). In addition, the NRC staff identified areas for which no independent calculations are necessary (see Attachment 1 to Matrix 10 of RS-001). In regard to the instrumentation and controls area (the example used in the comment), the NRC staff stated that independent calculations are not performed. However, for plants that do not have an NRC-approved setpoint methodology, a detailed review of the licensee's calculations for one instrument should be performed to verify that the licensee's application of the methodologies is correct and consistent with the NRC staff positions. The NRC staff has evaluated the comment as related to the guidance provided for the instrumentation and controls area and continues to believe that the approach presented in the draft review standard is appropriate for the NRC staff to reach a conclusion regarding acceptability of the licensee's calculations.
12	References NEI recommends that RS-001 include a stand-alone References section.	NEI March 31, 2003	For the most part, RS-001 refers to other documents for technical and process guidance and does not provide detailed technical or process guidance itself. Based on this, the staff does not believe that sufficient benefits exist for creating a separate references section for RS-001.
13	Future Revisions of RS-001 Because of the significant effort associated with an EPU application and the subsequent NRC staff review, NEI recommends that the initial use of RS-001 be monitored to identify "lessons learned" that can be incorporated into future revisions of the document.	NEI March 31, 2003	RS-001 is a living document and will be updated as needed to incorporate lessons learned and experience gained from power uprate reviews as well as other experience.
14	The document should be revisited and evaluated to determine if there is indeed a savings in review costs and RAIs issued.	STARS March 28, 2003	There are several goals for developing RS-001, including improving the consistency, effectiveness, efficiency, and documentation of the NRC staff's reviews of EPU applications. Future evaluations of and updates to RS-001 will consider all of the goals of the review standard.

NUMBER	COMMENT	SOURCE & DATE RECEIVED	RESPONSE
15	The document provides a draft Safety Evaluation (SE) for both boiling water reactors and pressurized water reactors. Section 1.3 of both SEs state, "The licensee's application for the proposed EPU follows the guidance in the Office of Nuclear Reactor Regulation's (NRR's) Review Standard (RS)-001, 'Review Standard for Extended Power Uprates,' to the extent that the review standard is consistent with the licensing basis of the plant." This statement invokes the review standard as guidance for the licensee. From a content perspective, this is not an issue. However, conceptually this establishes the review standard as a document similar to a Regulatory Guide or NUREG. The Review Standard does not have the same review, comment and publication requirements and controls. Where the current development and review has been extensive and comprehensive, there does not appear to be a requirement for future revisions to be as rigorous. Therefore, STARS recommends either striking that statement from the SE or formalizing the review and approval process to require public notification and comment.	STARS March 28, 2003	The NRC staff agrees. The NRC staff will develop an office instruction on the development of and revisions to review standards. The office instructions will provide appropriate thresholds for seeking public comment and for seeking endorsement from other stakeholders (e.g., Advisory Committee on Reactor Safeguards, Committee to Review Generic Requirements, Office of the General Counsel).
16	STARS is also concerned as to the issue of backfit. There is the potential that some of the criteria established by this review standard may pose issues of backfit on some licensees. The document includes provisions for criteria that do not apply to a licensee's licensing basis. It does not provide guidance on the issue criteria that could be considered backfit. STARS recommends some discussion of this topic be included.	STARS March 28, 2003	RS-001 encourages licensees to identify differences between their plant's licensing basis and the criteria in the review standard. The NRC staff plans to review a plant's power uprate application against the plant's licensing basis. Additional clarification related to this comment was added in RS-001 (See the Purpose section of RS-001). (This comment is similar to Comments 4, 7, and 10 above; and Comment 20 below.)
17	The draft review plan appears to require the development of a matrix to identify differences between the SRP and the licensing basis of the plant. ... It is suggested that the comparison be limited to areas that are of most interest to the NRC; specifically, those areas that are not bounded at the current power level or where a significant reduction in design margin may occur when the uprate is implemented.	Framatome ANP, Inc. May 2, 2003	The staff has identified the areas of interest for an extended power uprate in RS-001. The staff believes that to gain a sufficient level of understanding of the impacts of a proposed extended power uprate, a licensee should provide the information identified in the matrices in RS-001. Such information, for all areas of the scope of review, is necessary for the staff to determine if it agrees with the licensee's conclusions.
18	The matrix is supposed to include "other guidance," which includes regulatory guides and other documents that may not be part of the licensing basis. This requirement should be limited to those documents that are part of the licensing basis.	Framatome ANP, Inc. May 2, 2003	The matrices in RS-001 are generic. A licensee should clearly identify differences between their plant's licensing basis and the criteria in the review standard. In cases where the licensing basis is based on different criteria, the licensee should identify the criteria or provide a reference to the documents where the criteria exist. The staff plans to review the application against the plant's licensing basis.

NUMBER	COMMENT	SOURCE & DATE RECEIVED	RESPONSE
19	The draft review plan requires an audit of calculational files under certain conditions. Since the NRC always has this opportunity available, it seems unnecessary to require it. Our experience shows that this type of interaction places a significant burden on both the NRC and the applicant, a burden that appears unnecessary in this case. The intent of audits should be to ensure that methodologies are being properly applied rather than subjecting licensees to potential re-review of accepted methods.	Framatome ANP, Inc. May 2, 2003	The staff believes that providing guidance in RS-001 on when to perform such audits or calculations makes the review more consistent and transparent to stakeholders. The staff believes that audits and independent calculations, as defined by the criteria for independent calculations in the matrices in Section 2.2 of RS-001, are needed for the staff to gain the assurance it needs to complete its review. In general, the staff's review will be to ensure that methodologies are being properly applied. However, the staff notes that it may be necessary to revisit previously accepted methods to ensure that a proposed extended power uprate would not result in placing the plant's response outside of the applicability of the methods.
20	Several instructions are provided to the reviewers of the ECCS analysis that might not be consistent with plant-specific licensing bases. To avoid the potential for imposing unnecessarily stringent acceptance criteria, the basis for determining adequate safety should be the existing licensing basis.	Framatome ANP, Inc. May 2, 2003	RS-001 encourages licensees to identify differences between their plant's licensing basis and the criteria in the review standard. The NRC staff plans to review a plant's power uprate application against the plant's licensing basis. Additional clarification related to this comment was added in RS-001 (See the Purpose section of RS-001). (This comment is similar to Comments 4, 7, 10, and 16 above.)
21	The requirement to review "training for non-licensed plant staff" does not appear pertinent. Any plant modification may require some specialized training. So it is not clear why this particular instruction is included here.	Framatome ANP, Inc. May 2, 2003	This instruction is needed for the staff to confirm that the licensee has considered impacts of the extended power uprate on operations as well as other support staff at the plant.
22	An even more important action than formalizing a standard review plan is to establish a standard format for applications for extended power uprates. A standard format would convey much the same message and would facilitate both the development of the application and its review by the NRC.	Framatome ANP, Inc. May 2, 2003	The staff believes that RS-001 could be used by industry to guide its development of such a format.

NEI = Nuclear Energy Institute
STARS = Strategic Teaming and Resource Sharing

ATTACHMENT 3

Comments by The Advisory Committee on Reactor Safeguards on Past Extended Power Uprates

COMMENTS BY THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS ON PAST EXTENDED POWER UPRATES (EPUs)

NUMBER	COMMENT	REFERENCE	RESPONSE
1	The staff's safety evaluations do not reflect the substantial effort that went into the staff's review, including audits conducted onsite and at vendor facilities. The staff should develop improved guidance on the detail to be provided in safety evaluations. The staff should upgrade the safety evaluations to better reflect the depth and breadth of the staff's engineering evaluations.	Letters on Duane Arnold, Dresden, and Quad Cities EPUs October 17, 2001 and December 12, 2001	As part of the development of the Review Standard (RS)-001, "Review Standard for Extended Power Uprates," the NRC staff developed template safety evaluations that meet the guidance in NRR Office Instruction LIC-101, "License Amendment Review Procedures." The use of these template safety evaluations in future EPU reviews should enhance the NRC staff's documentation of its reviews. (See Sections 3.1 through 3.3 of RS-001)
2	It is important that the staff reviewing the power uprate application have a good process that communicates the importance of the [flow-assisted corrosion] monitoring program to the staff who inspect the uprated plant.	Letter on Clinton EPU March 14, 2002	The NRC staff has developed an inspection procedure for EPUs. In addition, the template safety evaluations in RS-001 provide guidance for reviewers to identify areas that are determined through the technical review to be good candidates for inspection. (See IP 71004, Section 5.0 of template safety evaluation in Sections 3.2 and 3.3 of RS-001, and Section 4.1 of RS-001)
3	Staff evaluations could have benefitted by including the results of independent computations and detailed checks of calculations to support the staff's review and audits of the procedures and conclusions described by the applicant. The staff should develop criteria for when independent assessments should be performed to complement its reviews of applicant submittals.	Letters on Duane Arnold and Brunswick EPUs October 17, 2001 and May 10, 2002	As part of the development of RS-001, the NRC staff developed guidelines for when and what type of independent analyses should be performed to supplement its review of licensees' EPU applications. (See Attachment 1 to each matrix in RS-001)
4	The process used by the staff and the Applicant was comprehensive enough to identify the important issues associated with pressurized water reactor power uprates. The process would be greatly improved by the availability of a standard review plan section to guide future reviews. A standard review plan would also clarify to both the public and licensees what is required for an application for power uprate to be found acceptable.	Letters on Dresden, Quad Cities and ANO-2 EPUs December 12, 2001 and March 14, 2002	The NRC staff has developed RS-001, which is consistent with and goes beyond the recommendation to develop an SRP section for power uprates.

NUMBER	COMMENT	REFERENCE	RESPONSE
5	Reduction in some of the times available for operator actions because of higher decay heat is especially significant for a power uprate.	Letter on Duane Arnold EPU October 17, 2001	The NRC staff will continue to review operator actions times related to safety significant scenarios as part of its reviews of EPU applications. (See Matrix 11 and Attachment 2 to Matrix 13 in Section 2 of RS-001)
6	Since integral tests of the plants' response can reveal otherwise undetected latent flaws, these tests should be conducted to confirm that these programs have achieved the desired result.	Letter on Dresden and Quad Cities EPUs December 12, 2001	The NRC staff has issued a draft standard review plan section on the testing programs for power uprates. This standard review plan section was issued for interim use and public comment and will be used to review licensee proposals for testing associated with EPUs. (See SRP 14.2.1 and Matrix 12 in Section 2 of RS-001)
7	We have not found a value for large transient tests that are commensurate with costs and risks and, therefore, support the position not to conduct the large-transient tests.	Letter on Clinton EPU March 14, 2002	The NRC staff has issued a draft standard review plan section on the testing programs for power uprates. This standard review plan section was issued for interim use and public comment and will be used to review licensee proposals for testing associated with EPUs. (See SRP 14.2.1 and Matrix 12 in Section 2 of RS-001)
8	Irradiation-assisted stress corrosion cracking (IASCC) of reactor internals is especially significant for a power uprate.	Letter on Duane Arnold EPU October 17, 2001	The NRC staff will continue to review the effect of EPUs on IASCC of reactor internals. (See Matrix 1 in Section 2 of RS-001)
9	Flow-assisted corrosion (FAC) is especially significant for a power uprate.	Letters on Duane Arnold and Clinton EPUs October 17, 2001 and March 14, 2002	The NRC staff will continue to review the effect of EPUs on FAC. (See Matrix 1 in Section 2 of RS-001)
10	Fatigue of feedwater piping is especially significant for a power uprate.	Letter on Duane Arnold EPU October 17, 2001	The NRC staff will continue to review the effect of EPUs on fatigue of feedwater piping. (See Matrix 2 in Section 2 of RS-001)
11	Containment response to accident events involving higher decay heat levels is especially significant for a power uprate.	Letter on Duane Arnold EPU October 17, 2001	The NRC staff will continue to review the effect of EPUs on containment response to accident events. (See Matrix 6 in Section 2 of RS-001)

NUMBER	COMMENT	REFERENCE	RESPONSE
12	Uncertainties in human reliability analysis are significant, but there is no mention of them in the staff's SE.	Letter on Brunswick EPUs May 10, 2002	The staff explicitly stated in the Brunswick safety evaluation that there may be large uncertainties associated with the various human reliability analysis (HRA) methodologies and thus, the absolute values cannot be used as the sole basis for determining the acceptability of a license application. However, the evaluations, if applied properly and consistently, can provide insights into the relative importance (or change in importance) of selected operator actions, can be used to focus the staff review of the license application on those aspects impacted by the EPU, and can be used to evaluate the overall relative change in risk due to the implementation of the EPU.

NUMBER	COMMENT	REFERENCE	RESPONSE
13	The applicant used human reliability models that have not been reviewed by the staff.	Letter on Brunswick EPUs May 10, 2002	The staff recognizes that none of the numerous, different HRA methodologies that are employed in probabilistic risk assessments (PRAs) throughout the industry have been formally reviewed and approved by the NRC. However, the NRC staff are familiar with the HRA methodologies used in the EPU license application, which are among the methods that comprise the current state-of-the-art. The staff recognizes that the HRA methodologies are evolving and that no particular HRA method may have a full consensus within the technical community. However, their use by a relatively large number of licensee PRA staff and by a number of PRA consultants, who use them to provide a means to estimate HEP values in a relatively coherent way that recognizes the influence of some situational characteristics (e.g., operator available response time), indicates their acceptance by these practitioners as the best available methods. Also, just because this is an area of active research does not invalidate the use of the current state-of-the-art methodologies to produce risk insights. As better and more refined methods are produced and/or evolve, the staff expects licensees to incorporate them into their PRAs. A paragraph was included in the Brunswick EPU safety evaluation which clarifies that the cited HRA methodologies have not been formally reviewed and approved by the NRC, but that they can be useful in the evaluation as discussed above.

NUMBER	COMMENT	REFERENCE	RESPONSE
14	The potential increases in the change in core damage frequency (ΔCDF), that could arise if the PRA were capable of modeling the effect of margin reductions on risk, are not included.	Letter on Brunswick EPUs May 10, 2002	The PRA can and does address the effects of margins reductions through the system/equipment success criteria evaluations. EPUs tend to reduce the margins that are traditionally identified in the deterministic, licensing arena (e.g., design basis accidents calculations). However, these margins reductions do not typically have a significant impact in the risk analyses, at least partially due to equipment additional capability. Based on the staff's experience during previous EPU reviews, there are typically no changes in system/equipment success criteria, except possibly the number of safety relief valves required to open during an event or as part of the automatic depressurization system. Thus, the results of EPU risk evaluations are consistent with and confirm the staff's expectation that the licensee can reduce deterministically-calculated margins without significantly impacting risk, which is reflected in the minimal impact EPUs have on system/equipment success criteria.

NUMBER	COMMENT	REFERENCE	RESPONSE
15	By not raising concerns about the quality of ΔCDF numbers, the staff implies some degree of acceptance. Maintaining public confidence is not served by tacit acceptance of unreviewed models. Improvements in PRA quality may be discouraged by making decisions, such as granting power uprates, by "accepting" PRAs without criticism because the application is not risk-informed. This review demonstrates an inherent problem in the "two-tier" regulatory system. The application for the EPU was not risk-informed, yet a PRA was submitted. This creates a situation in which the PRA is not seriously reviewed, although it is part of the record.	Letter on Brunswick EPUs May 10, 2002	Though EPU license applications are not designated as "risk-informed" submittals, licensees do submit risk information because it is identified as information that should be submitted in support of an EPU required by the NRC-approved General Electric (GE) topical reports NEDC-32424P-A (referred to as ELTR-1) and NEDC-33004P-A. This risk information is carefully and thoroughly reviewed by the staff, with the primary focus being to determine if special circumstances exist per Appendix D of Standard Review Plan (SRP) Chapter 19 (and RIS 2001-02) and to ensure that the information submitted is reasonable. This reasonableness check involves ensuring that the information meets the staff's expectations as identified by the staff's position and/or safety evaluation on the GE topical reports and reflects the experiences gained by the staff during previous EPU license application reviews. Thus, even though EPUs are not risk-informed, the staff does review the risk information provided by the licensee, consistent with the current guidance. Though the staff did identify some issues with the licensee's risk evaluation for the Brunswick application, none of these issues constituted the special circumstances that would raise a question regarding adequate protection and invoke the process identified in SRP Chapter 19 Appendix D, which would include NRR management notification and agreement prior to conducting a further more detailed review of the risks associated with the EPU. This finding was documented and used in the decisionmaking process that resulted in the approval of the application.

NUMBER	COMMENT	REFERENCE	RESPONSE
16	I do not think that the staff should accept results that are produced from [HEP] methodologies that are neither approved by the NRC, nor widely accepted.	Letter on ANO-2 EPU March 14, 2002	The staff recognizes that none of the numerous, different HRA methodologies that are employed in PRAs throughout the industry have been formally reviewed and approved by the NRC. However, the NRC staff are familiar with the HRA methodologies used in the EPU license application, which are among the methods that comprise the current state-of-the-art. The staff recognizes that the HRA methodologies are evolving and that no particular HRA method may have a full consensus within the technical community. However, the staff does not fully agree with the comment that these methodologies are not widely accepted by the technical community. Their use by a relatively large number of licensee PRA staff and by a number of PRA consultants, who use them to provide a means to estimate HEP values in a relatively coherent way that recognizes the influence of some situational characteristics (e.g., operator available response time), indicates their acceptance by these practitioners as the best available methods. Also, just because this is an area of active research does not invalidate the use of the current state-of-the-art methodologies to produce risk insights. As better and more refined methods are produced and/or evolve, the staff expects licensees to incorporate them into their PRAs. The staff review of the licensee's HRA methodologies, which were used in their risk impact evaluation, was to ensure that the licensee properly used the methodologies that they cited. A paragraph was added to the Brunswick EPU safety evaluation which clarifies that the cited HRA methodologies have not been formally reviewed and approved by the NRC, but that they can be useful in the risk evaluation as discussed above.

NUMBER	COMMENT	REFERENCE	RESPONSE
17	The Brunswick PRA submittal reports a LERF value of 4.27×10^{-6}/yr and a ΔLERF of about 2×10^{-7}/yr as a result of the power uprate, not including the SLCS modifications. The claim is that these values place this change to the licensing basis into Region II of the Regulatory Guide 1.174 acceptance guideline, which would permit this proposed power uprate. There are a number of things wrong with this view of Regulatory Guide 1.174. 1. The PRA did not include fire, seismic, or shutdown conditions. If included, these are likely to increase the assessed LERF value by a factor of 2. 2. There are two units on the site. As LERF is a site criterion that is a surrogate for the Commission's prompt fatality safety goal, then the LERF value for each unit must be added together to constitute the appropriate Regulatory Guide 1.174 site LERF. This increases the LERF by a factor of 2. 3. The LERF value submitted was a "point estimate" It can be guessed that the actual mean can be at least a factor of 2 greater than this. 4. The site LERF acceptance value is supposed to be a surrogate for the Commission's prompt fatality safety goal. The power uprate, to a first approximation, will increase the fission product inventory by 15% and, if the dose/consequence model were linear, this would increase the prompt fatalities by 15%. To account for this, the calculated LERF for comparison with the acceptance criteria in Regulatory Guide 1.174 should be increased by 15%.	Letter on Brunswick EPUs May 10, 2002	The staff notes that there are on-going interactions between the ACRS and the staff on many of these RG 1.174 implementation/interpretation issues. While these interactions continue, the staff will continue to address the large early release frequency (LERF) acceptance guidelines on an individual unit basis. The staff will also continue to qualitatively (including using simplistic estimations) address unquantified factors (e.g., external events) consistent with RG 1.174. For the Brunswick EPU application, the modification to the SLC system was performed to improve its success criteria for EPU conditions and is one of many plant modifications that the licensee stated it would implement to achieve the full EPU. The fact that this modification directly addresses the main risk impact of the EPU, which is from anticipated transient without scram (ATWS) events, demonstrates the integral part and benefit of this modification to the EPU license application. Section 2.2.5.4 of RG 1.174 states that "...if there is an indication that the CDF or LERF could considerably exceed 10^{-4} and 10^{-5}, respectively, in order for the change to be considered, the licensee may be required to present arguments as to why steps should not be taken to reduce CDF or LERF. Such an indication would result, for example, if (1) the contribution to CDF or LERF calculated from a limited scope analysis, such as the IPE or the IPEEE, significantly exceeds 10^{-4} and 10^{-5}, respectively ..." ***[CONTINUED ON NEXT PAGE]***

NUMBER	COMMENT	REFERENCE	RESPONSE
			[CONTINUED FROM PREVIOUS PAGE] This section further states, "[c]onsistent with the viewpoint that the guidelines are not to be used prescriptively, even if calculated ΔCDF and ΔLERF values are such that they place the change in Region I or II, it may be possible to make a case that the application should be treated as if it were in Region II or III if, for example, it is shown that there are unquantified benefits that are not reflected in the quantitative risk results ... In addition, if compensatory measures are proposed to counter the impact of the major risk contributors, even though the impact of these measures may not be estimated numerically, such arguments will be considered in the decision process." Crediting the SLC system modification as a compensatory measure, results in the plant's overall risk decreasing. From a risk perspective, the plant will be safer at EPU conditions with the SLC system modification implemented, than at the pre-EPU plant conditions without the SLC modifications.
18	Because of the significant changes to the physical plant and to the analytical models used to analyze the plant under accident conditions, the staff should review transition reload safety analyses to ensure that applicants properly incorporated plant design changes and parameters that describe the characteristics of the transition reload.	Letter on ANO-2 EPU March 14, 2002	The staff developed independent calculation and analysis criteria in Attachment 1 to Matrix 8 of RS-001. The staff will use this information in the decision making process for performing audits of the core reload analyses.
19	Although the plant reload analyses used for the uprates are based on methodology that has been reviewed and approved by the staff, we support the staff's continuing effort to audit them. We encourage staff audits of the application of reload analysis methods to transitional reloads for plants undergoing substantial power uprates.	Letter on GE CPPU Topical Report April 17, 2002	The staff will continue to audit analyses supporting EPU applications. These audits may also be extended to amendments requesting additional changes in the licensed operating conditions for plants that have already implemented EPUs. For EPUs involving transition cores, the staff identified additional analyses that need to be included in the EPU application. Audits of EPU applications involving transitional cores would also include review of topics that are unique to transitional cores.

NUMBER	COMMENT	REFERENCE	RESPONSE
20	We encourage the staff to continue to pay close attention to the details of core reload analyses at Brunswick and other BWR EPU plants. This is particularly important with regard to the ways that core thermal success criteria will continue to be met as more sophisticated fuel design and reload management techniques are implemented.	Letter on Brunswick EPUs May 10, 2002	The EPU audit plan includes review of some aspects of the fuel and core performance. The staff will also perform confirmatory analyses using EPU core designs (when appropriate).
21	The staff should assess the need for more detailed thermal-hydraulic models of the core, replacing the current "averaging" approaches, to complement present neutronic analyses that model the wide variations in fuel composition and power level throughout the core.	Letter on Brunswick EPUs May 10, 2002	Unless there is a safety concern that the current NRC-approved methodology can no longer provide reliable results, the NRC can only encourage licensees to update the capabilities of their computational techniques and codes. However, due to the core designs necessary to achieve EPUs, longer cycle length, or expanded operating domain, licensees are opting to use codes that have better and more detailed thermal-hydraulic and neutronic modeling capabilities in order to gain more thermal limits margin.
22	Susceptibility of the core to local power oscillations should be considered.	Letter on Duane Arnold EPU October 17, 2001	The staff evaluates the susceptibility of EPU core designs to the potential for power oscillations.
23	Susceptibility of the plant to ATWS and ATWS recovery are especially significant for a power uprate. In addition, effects of flattened power profiles on ATWS recovery methods should be addressed.	Letter on Duane Arnold EPU October 17, 2001	The staff evaluates the susceptibility of EPU cores to power oscillations and the effectiveness of the mitigation strategies to the oscillations (see question above).

ATTACHMENT 4

SRP Section 13.2.1
"Reactor Operator Training"

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

13.2.1 REACTOR OPERATOR TRAINING

REVIEW RESPONSIBILITIES

Primary - Equipment and Human Performance Branch (IEHB)

Secondary - None

I. AREAS OF REVIEW

The applicant's licensed operator training program, as described in the safety analysis report (SAR), is reviewed. This section of the SAR should contain the description and scheduling of the training program for reactor operators and senior reactor operators. The licensed operator training program also includes the requalification program as required in 10 CFR 50.54(i)(i-1) and 55.59.

A. Construction Permit (CP) and Early Stage Combined License (COL)

The training program descriptions should contain the following elements:

1. A description of the proposed training program including the subject matter of each initial licensed operator training course, the duration of the course (approximate number of weeks personnel are in full time attendance), the organization teaching the course or supervising instruction, and the titles of the positions for which the course is given. The program descriptions should include a chart showing the proposed schedule for licensing personnel prior to criticality. The schedule should be relative to expected fuel loading and should display the preoperational test period. The submittal should contain a commitment to conduct formal licensed operator, on-the-job training, and simulator training before initial fuel load.

2. The subjects covered in the training programs should include, as a minimum, those contained in 10 CFR 55.31, 55.41, 55.43, 55.45, and Regulatory Guide 1.8 for reactor operators and senior reactor operators as appropriate. The training program should also include provisions for upgrading reactor operator licenses and for licensing senior reactor operators who have not been licensed as reactor operators per Regulatory Guide 1.8. The training should be based on use of the systems approach to training (SAT) as defined in 10 CFR 55.4

3. The licensed operator requalification program should include the content described in 10 CFR 55.59 or should be based on the use of a systems approach to training as defined in 10 CFR 55.4.

4. Applicants should describe their program for providing simulator capability for their plants as described in 10 CFR 55.31, 55.45, 55.46, 50.34(f)(2)(i), and Regulatory Guide 1.149. In addition, the applicant should describe how it will ensure that its proposed simulator will correctly model its control room. Applicants should submit, prior to issuance of construction permits or other submittals, a general discussion of how the requirements will be met. Sufficient details should be presented to provide reasonable assurance that the requirements will be implemented prior to the issuance of a license.

5. The means for evaluating training program effectiveness for all licensed operators, in accordance with a systems approach to training.

B. Operating License (OL) or Late Stage Combined License (COL)

The training program descriptions should include the following elements:

1. The licensed operator training program descriptions should delineate clearly the extent to which the training program was accomplished at the approximate time of submittal of the SAR. Contingency plans for additional training for individuals to be licensed prior to criticality should be described in the event fuel loading is subsequently delayed from the date indicated in the SAR.

2. Reactor operations training at nuclear power plant simulation facilities that comply with Regulatory Guide 1.149. The applicant should provide the details of the program for simulator training, including length of time (weeks) and a description of the simulation facility as required by 10 CFR 55.45(b) and 55.46. The applicant should also provide details of the program to meet experience requirements for applicants for operator and senior operator licenses as required by 10 CFR 55.31 and 55.46.

3. The SAR should describe the applicant's plans for requalification training for licensed operators and senior operators.

   a. The subject matter of each course, including a syllabus or equivalent course description, the duration of the course (approximate number of weeks personnel are in full-time attendance), the organization teaching the course or supervising instruction, and the titles of the positions for which the course is given. The program should distinguish between classroom, on-the-job, and simulator training, before and after the initial fuel loading. It should include provisions for training on modifications to plant systems or functions.

The organization teaching the course or supervising the instruction and the qualifications of the instructors in the training program should be provided.

The subjects covered should include, as a minimum, those contained in 10 CFR 55.41, 55.43, 55.45, and Regulatory Guide 1.8 for reactor operators and senior reactor operators as appropriate. The training program should also include provisions for upgrading reactor operator licenses and for licensing senior reactor operators who have not been licensed as reactor operators per Regulatory Guide 1.8. The training should be based on the use of SAT as defined in 10 CFR 55.4.

b. The licensed operator requalification program should include the content described in 10 CFR 55.59 or be based on the use of a systems approach to training as defined in 10 CFR 55.4.

c. The means for evaluating training program effectiveness for all licensed operators, in accordance with SAT as defined in 10 CFR 55.4.

II. ACCEPTANCE CRITERIA

A. General Guidance

The SAR should demonstrate that the training provided, or to be provided, for reactor operators and senior reactor operators will be adequate to provide assurance that all reactor operator qualification requirement items will be met at the time needed, i.e., prior to operator license examinations, prior to fuel loading, or prior to appointment or reappointment to the position.

Criteria for acceptability, as they relate to licensed operator training and retraining programs, are:

1. The training and qualification requirements and guidance set forth in the following regulations and regulatory guides should be met or acceptable alternatives should be presented:

    a. 10 CFR Part 50, Section 50.54, items i through m

    b. 10 CFR Part 55, Sections 55.4, 55.31, 55.41, 55.43, 55.45, 55.46 and 55.59

    c. 10 CFR 50.34(f)(2)(i)

    d. Regulatory Guide 1.8, "Qualification and Training of Personnel for Nuclear Power Plants"

    e. Regulatory Guide 1.149, "Nuclear Power Plant Simulation Facilities for Use in Operator Training and License Examinations"

    f. NUREG-0711, "Human Factors Engineering Program Review Model"

    g. NUREG-1021, "Operator Licensing Examination Standards for Power Reactors"

2. Training programs shall be developed, established, implemented, and maintained using a systems approach to training as defined by 10 CFR 55.4. Training program development will be evaluated by the staff using the guidance in NUREG-0711 and training program content, and effectiveness will evaluated using NUREG-1220.

3. Formal segments of the initial licensed operator training program should be substantially completed when the preoperational test program begins.

4. The number of persons trained in preparation for licensed operator and senior operator licensing examinations prior to criticality should be sufficient to ensure that applicable regulatory requirements with respect to shift staffing can be met from the time of initial fuel loading, with allowances for examination contingencies and the need to avoid planned overtime.

5. The licensed operator requalification training program should adequately implement the requirements of 10 CFR 55.59.

B. Technical Rationale

The technical rationale for application of these acceptance criteria to reviewing licensed operator training is discussed in the following paragraphs:

1. Compliance with the relevant requirements of 10 CFR 50.54 items i through m requires the licensee to have licensed operators or senior operators present at the controls and responsible for manipulation of the controls or directing the licensed activities of other licensed operators, as appropriate.

   The reactor operator and senior reactor operator training programs, including initial and requalification training, established by the applicant provide the means to train individuals in the knowledge, skills, and abilities needed to perform licensed operator duties.

   Meeting these requirements provides assurance that only trained and qualified individuals will be licensed and assigned to carry out or direct operational activities, including manipulation of the controls and other activities affecting reactivity or power level.

2. Compliance with the relevant requirements of 10 CFR 55.4, 55.31, 55.41, 55.43, 55.45, 55.46, and 55.59 requires that the applicant for an operator's license and for requalification successfully complete written and operating examinations which demonstrate that the applicant possesses the knowledge, skills, and abilities needed to perform licensed activities.

   The reactor operator and senior reactor operator training programs, including initial and requalification training, established by the applicant provide the means to train individuals in the knowledge, skills, and abilities needed to perform licensed operator duties.

   Meeting these requirements provides assurance that only trained and qualified licensed individuals possessing the required knowledge, skills, and abilities will be assigned to, and conduct, licensed activities.

III. REVIEW PROCEDURES

Preparation for the review of Section 13.2.1 of the SAR should include familiarization with 10 CFR 50.54 items i through m; 10 CFR Part 55, Sections 55.4, 55.31, 55.41, 55.43, 55.45, 55.46, and 55.59; 10 CFR 50.34(f)(2)(i); Regulatory Guides 1.8 and 1.149; and NUREGs-0711 and -1021.

The reviewer should ensure that whenever the applicant has committed to follow the position of a regulatory guide, industry standard, or other reference document, the specific revision being referred to is identified. Similarly, whenever the reviewer is using a position in a reference document as a basis for acceptability, the revision being used should be identified.

The reviewer then determines, based upon the foregoing, the overall acceptability of the applicant's licensed operator training plans.

For standard design certification reviews under 10 CFR Part 52, the procedures above should be followed, as modified by the procedures in SRP Section 14.3 (proposed), to verify that the design set forth in the standard safety analysis report, including inspections, tests, analyses, and acceptance criteria (ITAAC), site interface requirements, and combined license action items, meets the acceptance criteria given in Subsection II. SRP Section 14.3 (proposed) contains procedures for the review of certified design material (CDM) for the standard design, including the site parameters, interface criteria, and ITAAC.

IV. EVALUATION FINDINGS

The reviewer should verify that the information presented in the review supports an evaluation finding statement of the following type, to be used in the staff's safety evaluation report:

> The staff concludes that the training program for licensed operators and senior operators is acceptable and meets the requirements of 10 CFR 50.54 items i through m and 10 CFR Part 55, Sections 55.4, 55.31, 55.41, 55.43, 55.45, 55.46, and 55.59. This conclusion is based on the following:

For Construction Permit (CP) or Early Stage Combined License (COL)

The overall conduct and administration of the licensed operator training programs is the responsibility of the Plant Manager. The Training Manager is responsible for development, implementation, evaluation, and documentation of the licensed operator training programs.

The applicant states that a training program will be established to provide licensed operators with sufficient knowledge and operating experience to start up, operate, and maintain the plant in a safe manner. The licensed operator training program, derived from a systems approach to training, is to be developed by the applicant and will meet the regulatory guidance of Regulatory Guide 1.8. Licensed operators and senior operators will receive training in security procedures, radiological emergency plans, administrative procedures, and radiation protection. Simulation facilities used for licensed operator training program should meet the guidance of Regulatory Guide 1.149.

The information submitted relative to these subjects is satisfactory for the preoperational test program, for operator licensing, and for fuel loading.

For Operating License (OL) or Late Stage Combined License (COL)

The overall conduct and administration of the licensed operator training program is the responsibility of the Plant Manager. The Training Manager, reporting to the Plant Manager, is responsible for administering the licensed operator training program and monitoring program effectiveness. The applicant states that the licensed operator training program will provide reasonable assurance that decisions and actions by licensed operators and senior operators during all plant conditions will be made consistent with plant safety procedures and operational limits established to protect the public health and safety. The licensed operator training program has been designed to meet the individual needs of the participants, depending upon their backgrounds, previous training, and expected job assignment. The program will meet the guidelines of Regulatory Guide 1.8 and 10 CFR Part 55. Simulation facilities used in the training program shall meet the requirements of 10 CFR 55.31, 55.45(b), 55.46, and 50.34(f)(2)(1), and the guidelines of Regulatory Guide 1.149. Over [state specific number provided by the licensee] candidates will have completed the entire training program prior to the fuel loading so that a sufficient number of licensed operators should be available to meet the requirements of 10 CFR 50.54.

The licensed operator requalification training program conforms to the requirements of 10 CFR Part 50 and 10 CFR 55.59 and follows the guidance given in Regulatory Guide 1.8.

For Design Certification Reviews

For design certification reviews, the findings will also summarize, to the extent that the review is not discussed in other safety evaluation report sections, the staff's evaluation of inspections, tests, analyses, and acceptance criteria (ITAAC), including design acceptance criteria (DAC), site interface requirements, and combined license action items that are relevant to this SRP section.

V. IMPLEMENTATION

The following is intended to provide guidance to applicants and licensees regarding the NRC staff's plans for using this SRP section.

This SRP section will be used by the staff when performing safety evaluations of license applications submitted by applicants pursuant to 10 CFR Part 50 or 10 CFR Part 52. Except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method described herein will be used by the staff in its evaluation of conformance with Commission regulations.

The provisions of this SRP section apply to reviews of applications docketed 6 months or more after the date of issuance of this SRP section.

Implementation schedules for conformance to parts of the review plan discussed herein are contained in the referenced regulatory guides and NUREGS.

VI. REFERENCES

1. 10 CFR Part 50, "Licensing of Production and Utilization Facilities."
2. 10 CFR Part 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants."
3. 10 CFR Part 55, "Operators' Licenses."
4. Regulatory Guide 1.8, "Qualification and Training of Personnel for Nuclear Power Plants."
5. Regulatory Guide 1.149, "Nuclear Power Plant Simulation Facilities for Use in Operator Training and License Examinations."
6. NUREG-0711, "Human Factors Engineering Program Review Model."
7. NUREG-1021, "Operator Licensing Examination Standards for Power Reactors."
8. NUREG-1220, "Training Review Criteria and Procedures."

ATTACHMENT 5

SRP Section 13.2.2
"Training for Nonlicensed Plant Staff"

U.S. NUCLEAR REGULATORY COMMISSION
STANDARD REVIEW PLAN
OFFICE OF NUCLEAR REACTOR REGULATION

13.2.2 TRAINING FOR NONLICENSED PLANT STAFF

REVIEW RESPONSIBILITIES

Primary - Equipment and Human Performance Branch (IEHB)

Secondary - None

I. AREAS OF REVIEW

The applicant's training program for the nonlicensed plant staff, as described in the safety analysis report (SAR), is reviewed. This section of the SAR should contain the description and scheduling of the training and retraining programs for the nonlicensed plant staff.

A. Construction Permit (CP) and Early Stage Combined License (COL)

The program description should be for each position or organizational unit identified in SAR Section 13.1.2. The schedule is reviewed to verify that it is tied to expected fuel loading, reflects expected completion of required initial training prior to fuel load, and adequately covers the preoperational test period. The training program description should include the following elements:

1. The applicant's commitment to meet the guidelines of Regulatory Guide 1.8 for nonlicensed personnel.

2. For positions covered by 10 CFR 50.120, a commitment to meet the requirements of 10 CFR 50.120 at least 18 months before fuel load.

3. A commitment to conduct an onsite formal training program and on-the-job training such that the entire plant staff will be qualified before the initial fuel loading.

4. A commitment to conduct an initial fire protection training program including:
   a. Periodic drills during construction.
   b. Provisions for indoctrination of construction personnel, as necessary.

   The commitment to verify that initial fire protection training will be completed prior to receipt of fuel at the site.

5. The applicant's plans for conducting a position task analysis are reviewed to verify that the tasks performed by persons in each position are defined, and that the training, in conjunction with education and experience, is identified to provide assurance that the tasks can be effectively carried out.

6. For all plant personnel identified in SAR Section 13.1.2, the proposed subject matter of each course, the duration of the course (approximate number of weeks personnel are in full-time attendance), the organization teaching the course or supervising instruction, and the titles of the positions for which the course is given.

7. A description of the provisions for training employees and nonemployees whose assistance may be needed in a radiological emergency, as required by 10 CFR Part 50, Appendix E, Section II.F.

8. A description of the training program for the individual(s) responsible for the formulation and assurance of the implementation of the fire protection program. The training program description is reviewed to verify that it adequately addresses those items listed in Branch Technical Position SPLB 9.5-1 attached to Standard Review Plan (SRP) Section 9.5.1.

9. The proposed means for evaluating the training program effectiveness for all employees in accordance with the systems approach to training.

B. Operating License (OL) and Later Stage Combined License (COL)

The training program description is verified by identifying the extent to which the training program has been accomplished at the approximate time of the SAR submittal. The description verification includes, contingency plans for additional training in the event that fuel loading is significantly delayed from the date indicated in the SAR.

The applicant's plans for retraining of plant nonlicensed personnel are also reviewed and verified to adequately identify the additional plant staff categories for which retraining will be provided, and the nature, scope, and frequency of such retraining (13.2.2.2). The program should include provisions for training on modifications to plant systems or functions. The program description should include the following elements:

1. A detailed description of the training programs for nonlicensed personnel to meet the guidelines of Regulatory Guide 1.8.

2. A detailed description of the training programs developed using a systems approach to training, as defined in 10 CFR 55.4, for all positions covered by 10 CFR 50.120.

3. For programs not covered under 10 CFR 50.120, the subject matter of each course, including a syllabus or equivalent course description, the duration of the course (approximate number of weeks personnel are in full-time attendance), the organization teaching the course or supervising instruction, and the titles of the positions for which the course is given. The program is verified to distinguish between classroom training and on-the-job training, before and after the initial fuel loading.

   Any difference in the training programs for individuals based on the extent of previous nuclear power plant experience. The structuring of training based on experience groups is verified to appropriately address the following categories of personnel experience:
   a. Individuals with no previous experience.
   b. Individuals who have had nuclear experience at facilities not subject to licensing.
   c. Individuals who have had experience at comparable nuclear facilities.

4. A detailed description of the fire protection training and retraining for the initial plant staff and replacement personnel. The program is verified to adequately address:
   a. The training planned for each member of the fire brigade.
   b. The type and frequency of periodic firefighting drills.
   c. The training provided for all remaining staff members, including personnel responsible for maintenance and inspection of fire protection equipment.
   d. The indoctrination and training provided for people temporarily assigned onsite duties during shutdown and maintenance outages, particularly those allowed unescorted access.
   e. The training provided for the fire protection staff members. The program description is verified to include the course of instruction, the number of hours of each course, and the organization conducting the training.

5. OL and COL applicants should provide a description of the results of the position's task analysis and the program as implemented. The description is reviewed to verify that the program has been implemented based on the plans provided previously.

6. A description of training and exercises, via periodic drills, of radiation emergency plans required by 10 CFR Part 50, Appendix E, Section IV.F. The training program is verified to include initial training and periodic retraining for categories of employees and nonemployees whose assistance may be needed in the event of a radiological emergency.

7. Means for evaluating the training program effectiveness for each employee in accordance with a systems approach to training.

C. Review Interfaces

The primary human performance review branch performs the following reviews under the SRP sections indicated:

SRP Sections 13.1.1 through 13.1.3 - Conduct of Operations,
SRP Section 13.2.1 - Reactor Operator Training,
SRP Section 13.5.2.1 - Administrative Procedures - General,
SRP Section 13.5.2.2 - Operating and Emergency Operating Procedures,
SRP Section 18.0 - Human Factors Engineering,

The primary human performance review branch will coordinate evaluations and reviews by other branches that support the overall review of training requirements for nonlicensed plant staff as follows:

1. With the branch responsible for Emergency Preparedness and Radiation Protection, as part of its primary review responsibility for SRP Section 13.3, 10 CFR Part 50, Appendix E, Sections II.F and IV.F, as they relate to training of personnel used during emergencies. Additionally, as part of its primary review responsibilities for SRP Section 12.5, 10 CFR 19.12 as it relates to radiological protection training,

2. With the office responsible for Safeguards as part of its primary review responsibility for SRP Section 13.6 for training of personnel controlling secured areas.

3. With the branch responsible for Plant Systems as part of its primary review responsibility for SRP Section 9.5.1 for fire protection training.

For those areas of review identified above as being part of the review under other SRP sections, the acceptance criteria necessary for the review and their methods of application are contained in the referenced SRP sections.

Paperwork Reduction Act Statement

The information collections contained in this NUREG are covered by the requirements of 10 CFR Parts 50, 52, 55, 19, and 26 which were approved by the Office of Management and Budget, approval numbers 3150-0011, 0151, 0018, 0044, and 0146.

Public Protection Notification

If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

II. ACCEPTANCE CRITERIA

A. General Guidance

The SAR should demonstrate that the training provided, or to be provided, for each position on the plant staff will be adequate to provide assurance that all plant staff personnel training and qualification requirements will be met at the time needed, i.e., prior to preoperational tests, prior to fuel loading, or prior to appointment or reappointment to the position.

Staff acceptance criteria in this subsection are designed to provide reasonable assurance that an applicant in compliance with these criteria will meet the relevant requirements of the following regulations:

1. 10 CFR 19.12 as it relates to appropriately informing and instructing personnel regarding the presence of radioactive materials and radiation, health protection problems associated with exposure thereto, means and responsibilities for protection of workers therefrom, and the availability upon request of radiation exposure reports. The personnel that must be so informed and instructed are all individuals who are likely to receive in a year an occupational dose greater than 1 mSv (100 mrem).

2. 10 CFR 26.21 and 10 CFR 26.22 as they relate to providing personnel training in conjunction with the fitness-for-duty program.

3. 10 CFR 50.34(a) and (b) as they relate to details of training given to nonlicensed plant personnel and a schedule for such training.

4. 10 CFR 50.40(b) as it relates to training being an integral part of personnel technical qualification, which contributes to the finding that the applicant is technically qualified to engage in licensing activities.

5. 10 CFR 50.120 and 10 CFR 52.78 as they relate to derivation of training programs from a systems approach to training.

6. 10 CFR Part 50, Appendix E, Sections II.F and IV.F, as they relate to establishing emergency preparedness training and retraining programs covering employees and other nonemployees whose assistance may be needed in a radiological emergency.

B. Specific Criteria

Specific criteria necessary to meet the relevant requirements of 10 CFR 19.12, 26.21, 26.22, 50.34(a) and (b), 50.40(b), 50.120, and 52.78 are as follows:

1. The nonlicensed plant personnel should be trained in accordance with an appropriate ANSI standard as endorsed by Regulatory Guide 1.8.

2. Training programs shall be developed, established, implemented, and maintained using a systems approach to training as required by 10 CFR 50.120 and 10 CFR 52.78 and as defined in 10 CFR 55.4. Training program development will be evaluated by the staff using the guidance contained in NUREG-0711 and training program content and effectiveness will be evaluated using NUREG-1220.

3. Simulation facilities used for training nonlicensed plant personnel should meet the guidelines of Regulatory Guide 1.149.

4. Personnel to be granted access to protected areas or to emergency operations facilities shall be trained to ensure understanding of information related to the fitness-for-duty program, including the associated policies and procedures, the hazards and effects associated with drugs and alcohol, available employee assistance programs, responsibilities under the policy, and the consequences that may result from lack of adherence to the policy, as required in 10 CFR 26.21. Managers, supervisors, and persons assigned to escort duties must be trained to ensure they understand the roles and responsibilities of personnel involved in the fitness-for-duty program, techniques for recognizing drugs and indications of drug possession or use, techniques for behavioral observation, and procedures for initiating corrective actions under the program, as required in 10 CFR 26.22.

5. Training programs related to radiological emergencies shall meet the requirements of 10 CFR 50, Appendix E, Section II.F or IV.F, as applicable. The detailed evaluation criteria and methods for the verification of overall compliance with these requirements are contained in SRP Section 13.3.

6. Formal segments of the initial training program should be substantially completed when the preoperational test program begins.

7. The number of people for whom training is planned prior to fuel load should be sufficient to ensure that applicable technical specification conditions with respect to the number of plant personnel can be met from the time of initial fuel loading of the first unit, with due allowance given for contingencies and the need to avoid planned overtime for supervisory personnel during the startup phase.

8. Refresher training for nonlicensed personnel should be periodic and not less frequent than every 2 years and should include, at a minimum, refresher instruction on administrative, radiation protection, emergency, and security procedures.

9. The detailed guidance and criteria for review of radiological protection training and retraining programs, including the evaluation of their adequacy in informing and instructing personnel pursuant to the requirements of 10 CFR 19.12, is described in SRP Section 12.5.

10. Fire Protection Training

    a. <u>Fire Brigade Training</u>

The fire brigade training program shall in general follow the guidelines of Branch Technical Position (BTP) SPLB 9.5-1 to ensure that the capability to fight potential fires is established and maintained. The program shall consist of an initial classroom instruction program followed by periodic classroom instruction, firefighting practice, and fire drills as follows:

(1) Instruction

(a) The initial classroom instruction shall include:

i) Indoctrination in the plant firefighting plan with specific identification of each individual's responsibilities.

ii) Identification of the type and location of fire hazards and associated types of fires that could occur in the plant.

iii) The toxic and corrosive characteristics of expected products of combustion.

iv) Identification of the location of firefighting equipment for each fire area and familiarization with the layout of the plant, including access and egress routes to and from each area.

v) The proper use of available firefighting equipment and the correct method of fighting each type of fire. The types of fires covered should include fires in energized electrical equipment, fires in cables and cable trays, hydrogen fires, fires involving flammable and combustible liquids or hazardous process chemicals, fires resulting from construction or modifications (welding), and record file fires.

vi) The proper use of communication, lighting, ventilation, and emergency breathing equipment.

vii) The proper method for fighting fires inside buildings and confined spaces.

viii) The direction and coordination of the firefighting activities (fire brigade leaders only).

ix) Detailed review of firefighting strategies and procedures.

x) Review of the latest plant modifications and corresponding changes in firefighting plans.

Note--Items ix and x may be deleted from the training of no more than two of the nonoperations personnel who may be assigned to the fire brigade.

(b) The instruction shall be provided by qualified individuals who are knowledgeable, experienced, and suitably trained in fighting the types of fires that could occur in the plant and in using the types of equipment available in the nuclear power plant.

(c) Instruction shall be provided to all fire brigade members and fire brigade leaders.

(d) Regular planned meetings shall be held at least every 3 months for all brigade members to review changes in the fire protection program and other subjects as necessary.

(e) Periodic refresher training sessions shall be held to repeat the classroom instruction program for all brigade members over a 2-year period. These sessions may be concurrent with the regular planned meetings.

(2) Practice

Practice sessions shall be held for each shift fire brigade on the proper method of fighting the various types of fires that could occur in a nuclear power plant. These sessions shall provide brigade members with experience in actual fire extinguishment and the use of emergency breathing apparatus under strenuous conditions encountered in firefighting. These practice sessions shall be provided at least once per year for each fire brigade member.

(3) Drills

(a) Fire brigade drills shall be performed in the plant so that the fire brigade can practice as a team.

(b) Drills shall be performed at regular intervals not to exceed 3 months for each shift fire brigade. Each fire brigade member should participate in each drill, but must participate in at least two drills per year.

A sufficient number of these drills, but not less than one for each shift fire brigade per year, shall be unannounced to determine the firefighting readiness of the plant fire brigade, brigade leader, and fire protection systems and equipment. Persons planning and authorizing an unannounced drill shall ensure that the responding shift fire brigade members are not aware that a drill is being planned until it is begun. Unannounced drills shall not be scheduled more frequently than 4 weeks apart.

At least one drill per year shall be performed on a "back shift" for each shift fire brigade.

(c) The drills shall be pre-planned to establish the training objectives of the drill and shall be critiqued to determine how well the training objectives have been met.

Unannounced drills shall be planned and critiqued by members of the management staff responsible for plant safety and fire protection. Performance deficiencies of a fire brigade or of individual fire brigade members shall be remedied by scheduling additional training for the brigade or members. Unsatisfactory drill performance shall be followed by a repeat drill within 30 days.

(d) At 3-year intervals, a randomly selected unannounced drill shall be critiqued by qualified individuals independent of the licensee's staff. A written report from such individuals shall be available for NRC review.

(e) Drills shall, as a minimum, include the following:

i) Assessment of fire alarm effectiveness, time required to notify and assemble the fire brigade, selection, placement and use of equipment, and firefighting strategies.

ii) Assessment of each brigade member's knowledge of his or her role in the firefighting strategy for the area assumed to contain the fire. Assessment of the brigade member's compliance with established plant firefighting procedures and use of firefighting equipment, including self-contained emergency breathing apparatus, communication equipment, and ventilation equipment, to the extent practicable.

iii) The simulated use of firefighting equipment required to cope with the situation and type of fire selected for the drill. The area and type of fire chosen for the drill should differ from those used in the previous drill so that brigade members are trained in fighting fires in various plant areas. The situation selected should simulate the size and configuration of a fire that could reasonably occur in the area selected, allowing for fire development due to the time required to respond, to obtain equipment, and organize for the fire, assuming the loss of automatic suppression capability.

iv) Assessment of the brigade leader's direction of the firefighting effort as to thoroughness, accuracy, and effectiveness.

(4) <u>Records</u>

Individual records of training provided to each fire brigade member, including drill critiques, shall be maintained for at least 3 years to ensure that each member receives training in all parts of the training program. These records of training shall be available for NRC review. Retraining or broadened training for firefighting within buildings shall be scheduled for all those brigade members whose performance records show deficiencies.

b. <u>Fire Protection Staff</u>

Training for the fire protection staff members shall include courses in:

(1) Design and maintenance of fire detection, suppression, and extinguishing systems.

(2) Fire prevention techniques and procedures.

(3) Training and manual firefighting techniques and procedures for plant personnel and the fire brigade.

c. Other Station Employees

(1) Instruction

(a) Instruction shall be provided for all employees once a year. It shall be repeated on an annual basis. The instruction shall be given, as appropriate, on (i) the fire protection plan (ii) the evacuation routes, and (iii) the procedure for reporting a fire.

(b) Instruction shall be provided for security personnel that addresses (i) entry procedures for outside fire departments, (ii) crowd control for people exiting the station, and (iii) procedures for reporting potential fire hazards observed when touring the facility.

(c) Instruction should be provided to all shift personnel that complements that provided members of the fire brigade.

(d) Instruction shall be provided to temporary employees so that they are familiar with (i) evacuation signals, (ii) evacuation routes, and (iii) the procedure for reporting fires.

(2) Drills

All employees should participate in an annual evacuation drill.

C. Technical Rationale

The technical rationale for application of these acceptance criteria to reviewing nonlicensed plant staff training is discussed in the following paragraphs:

1. To comply with the relevant requirements of 10 CFR 19.12 the applicant must provide, to all individuals who are likely to receive in a year an occupational dose in excess of 1 mSv (100 mrem), information and instruction on the health effects of radiation and means to minimize exposure.

   The nonlicensed staff training program established by the applicant provides the means to train individuals in precautions and procedures to minimize radiation exposure.

Meeting these requirements provides assurance that the applicant will provide employees with the information needed to minimize radiation exposure.

2. To comply with the relevant requirements of 10 CFR 26.21 and 26.22, the applicant must provide initial and refresher training to ensure that plant staff understand the policy, procedures, and responsibilities of the applicant's fitness-for-duty program.

   The nonlicensed staff training program established by the applicant provides the means to train individuals in the policies, procedures, and responsibilities of the fitness-for-duty program. The fitness-for-duty program provides a means for ensuring that plant staff members understand their roles and responsibilities in having only fit individuals present and involved in plant activities.

   Meeting these requirements provides assurance that only trained and fit individuals will be on site and involved in plant activities.

3. To comply with the relevant requirements of 10 CFR 50.34(a) and (b) the applicant must submit an SAR, with at least the minimum information described in the requirements. Required information includes plans for training personnel and personnel qualification requirements.

   The nonlicensed staff training program established by the applicant provides the means to train individuals in the knowledge, skills, and abilities needed to perform required tasks, particularly those tasks associated with fire brigades or radiological response teams, where the skills are not used on a day-to-day basis.

   Meeting these requirements provides assurance that trained personnel will be available to perform needed tasks to ensure safe plant operation and response to emergency situations.

4. To comply with the relevant requirements of 10 CFR 50.40(b), the applicant must be technically qualified to engage in activities associated with the design, construction, and operation of a nuclear power plant in accordance with the regulations in 10 CFR Part 50.

   The nonlicensed staff training program established by the applicant provides the means to train individuals in the knowledge, skills, and abilities needed to perform required tasks, particularly those tasks associated with fire brigades or radiological response teams, where the skills are not used on a day-to-day basis. The applicant's plan and program for training of nonlicensed staff provides insight into the applicant's approach to safe plant operation. This information contributes to the determination that an applicant is technically qualified by ensuring that appropriate considerations were used in the establishment of general training and qualification requirements for all nonlicensed personnel.

   Meeting these requirements provides assurance that the applicant is technically qualified to engage in the proposed activities and has established the necessary training program to safely operate the proposed facility.

5. To comply with the requirements of 10 CFR 50.120 and 10 CFR 52.78, the training programs for specified categories of personnel, including several nonlicensed personnel categories, must be established, implemented, and maintained using a systems approach to training as defined in 10 CFR 55.4.

   The non-licensed staff training program established by the applicant provides the means to train individuals in the knowledge, skills, and abilities needed to perform required tasks.

   Meeting these requirements provides assurance that trained personnel will be available to perform needed tasks to ensure safe plant operation and appropriate response to abnormal or emergency situations.

III. REVIEW PROCEDURES

Preparation for the review of Section 13.2 of the SAR should include familiarization with the documents listed in Subsection VI of this SRP section.

The reviewer should ensure that, whenever the applicant has committed to follow the position of a regulatory guide, industry standard, or other reference document, the specific revision being referred to is identified. Similarly, whenever the reviewer is using a position in a reference document as a basis for acceptability, the revision being used should be identified.

The reviewer should also ensure that the applicant has committed to a reasonable implementation schedule for the training programs and that the schedule relates to the fuel loading date. The reviewer may consult with the branch with primary responsibility for fire protection for the review of fire protection training.

The reviewer then determines, based upon the foregoing, the overall acceptability of the applicant's plant staff training plans.

For standard design certification reviews under 10 CFR Part 52, the procedures above should be followed, as modified by the procedures in SRP Section 14.3, to verify that the design set forth in the standard safety analysis report, including inspections, tests, analyses, and acceptance criteria (ITAAC), site interface requirements, and combined license action items, meet the acceptance criteria given in Subsection II. SRP Section 14.3 contains procedures for the review of certified design material (CDM) for the standard design, including the site parameters, interface criteria, and ITAAC.

IV. EVALUATION FINDINGS

The reviewer should verify that the information presented and should ensure that the review supports an evaluation finding statement of the following type, to be used in the staff's safety evaluation report:

> The staff concludes that the training for nonlicensed plant staff personnel is acceptable and meets the requirements of 10 CFR 19.12; 26.21 and 26.22; 50.34 (a or b); 50.40(b); and 50.120. This conclusion is based on the following:

For Construction Permit (CP)Only

The applicant has described in the SAR, in accordance with the requirements of 10 CFR 50.34(a)(6), an acceptable preliminary plan for training of nonlicensed plant personnel and appropriate commitments with respect to the plan so that the plan has been demonstrated to satisfy relevant requirements as discussed further below.

For Operating License or Combined License (COL)

The applicant has described in the SAR, in accordance with the requirements of 10 CFR 50.34(b)(7), the details of its training program for nonlicensed personnel, including appropriate commitments with respect to the program, the training given to nonlicensed plant personnel, and a schedule for that training as related to the applicant's presently scheduled fuel load date.

The training and retraining of nonlicensed personnel meet the guidance of Regulatory Guide 1.8.

The applicant meets the requirements of 10 CFR 19.12 by having a training program that informs and instructs personnel regarding radioactive materials and radiation, health protection problems associated with exposure to radiation, the means and responsibilities for the protection of workers from radiation, and the availability upon request of radiation exposure reports. The findings regarding radiation protection training and retraining programs that address these issues in greater detail are presented in Section 12.5 of this report.

The applicant meets the requirements of 10 CFR 26.21 and 26.22 by having a training program to ensure that personnel are adequately informed regarding the fitness-for-duty policy. Supervisors and persons assigned to escort duties will be trained to ensure that they understand their roles, responsibilities, and procedures for the fitness-for-duty program. This training program will ensure that they will possess knowledge and skills necessary for recognition of behavioral changes, drugs, and/or indications of the use of drugs.

The applicant has committed to establish, implement, and maintain training programs that will utilize a systems approach to training as required by 10 CFR 50.120 and as defined in 10 CFR 55.4.

Fire brigade personnel will undergo classroom instruction, firefighting practice, and periodic fire drills.

The simulation facilities used in the training program meet the guidelines of Regulatory Guide 1.149.

The training program includes initial training and periodic retraining for categories of employees and nonemployees whose assistance may be needed in the event of a radiological emergency. The findings regarding the adequacy of training and retraining programs related to radiological emergencies are presented in Section 13.3 of this report.

All initial training of the nonlicensed plant staff is scheduled to be completed prior to fuel loading.

Meeting the staff's requirements given above provides an acceptable basis for finding that, insofar as the training of nonlicensed personnel is concerned, the applicant meets the technical qualification requirements of 10 CFR 50.40(b) of the Commission's regulations.

For design certification reviews, the findings will also summarize, to the extent that the review is not discussed in other safety evaluation report sections, the staff's evaluation of inspections, tests, analyses, and acceptance criteria (ITAAC), including design acceptance criteria (DAC), site interface requirements, and combined license action items that are relevant to this SRP section.

V. IMPLEMENTATION

The following is intended to provide guidance to applicants and licensees regarding the NRC staff's plans for using this SRP section.

This SRP section will be used by the staff when performing safety evaluations of license applications submitted by applicants pursuant to 10 CFR Part 50 or 10 CFR Part 52. Except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method described herein will be used by the staff in its evaluation of conformance with Commission regulations.

The provisions of this SRP section apply to the review of applications docketed 6 months or more after the date of issuance of this SRP section.

Implementation schedules for conformance to parts of the review method discussed herein are contained in the referenced regulatory guides and NUREGS.

VI. REFERENCES

1. 10 CFR Part 19, "Notices, Instructions and Reports to Workers: Inspections and Investigations."

2. 10 CFR Part 26, "Fitness For Duty Programs."

3. 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities."

4. 10 CFR Part 50, Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities."

5. 10 CFR Part 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants."

6. Regulatory Guide 1.8, "Qualification and Training of Personnel for Nuclear Power Plants."

7. Regulatory Guide 1.149, "Nuclear Power Plant Simulation Facilities for Use in Operator Training and License Examinations."

8. Branch Technical Position SPLB124 9.5-1, attached to SRP Section 9.5.1, "Fire Protection."

9. NUREG-0711, "Human Factors Engineering Program Review Model."

10. NUREG-1220, "Training Review Criteria and Procedures."

11. Generic Letter 86-04, "Policy Statement on Engineering Expertise on Shift," February 1986.

ATTACHMENT 6

SRP Section 13.5.2.1
“Operating and Emergency Operating Procedures”

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

13.5.2.1 OPERATING AND EMERGENCY OPERATING PROCEDURES

REVIEW RESPONSIBILITIES

Primary - Equipment and Human Performance Branch (IEHB)

Secondary - Reactor Systems Branch (SRXB), Plant Systems Branch (SPLB)

I. AREAS OF REVIEW

The staff reviews the applicant's plan for development and implementation of operating procedures as described in the applicant's safety analysis report (SAR). This section of the SAR should describe the operating procedures that will be used by the operating organization (plant staff) to ensure that routine operating, off-normal, and emergency activities are conducted in a safe manner. It is not expected that detailed written procedures will be included in the SAR. It is recognized that development of detailed procedures and associated training materials may be beyond the scope of the application (e.g., for design certification) and then would be the responsibility of a combined license (COL) applicant referencing the certified design. The SAR should provide descriptions of the content and development process for procedures as detailed below, including preliminary schedules for preparation of procedures.

A. Procedure Classification

The SAR or other submittal should describe the different classifications of procedures the operators will use in the control room and locally in the plant for plant operations. The group within the operating organization having the responsibility for maintaining the procedures should be identified and the general format and content of the different classifications should be described. It is not necessary that each applicant's procedures conform precisely to the same classification since the objective is to ensure that procedures will be available to the plant staff to accomplish the functions contained in the listing of Regulatory Guide 1.33. For example, some licensees prefer a classification of abnormal operating procedures, whereas others may use off-normal condition procedures. Examples of classifications follow:

1. System Procedures. Procedures that provide instructions for energizing, filling, venting, draining, starting up, shutting down, changing modes of operation, returning to service following testing (if not contained in the applicable testing procedure), and other instructions appropriate for operation of systems important to safety.

2. General Plant Procedures. Procedures that provide instructions for the integrated operations of the plant, e.g., startup, shutting down, shutdown, power operation and load changing, process monitoring, and fuel handling.

3. Off-Normal Condition Procedures. Procedures that specify operator actions for restoring an operating variable to its normal controlled value when it departs from its normal range or to restore normal operating conditions following a transient. Such actions are invoked following an operator observation or an annunciator alarm indicating a condition which, if not corrected, could degenerate into a condition requiring action under an emergency operating procedure (EOP).

4. Emergency Operating Procedures. Procedures that direct actions necessary for the operators to mitigate the consequences of transients and accidents that cause plant parameters to exceed reactor protection system or engineered safety features actuation setpoints.

5. Alarm Procedures. Procedures that guide operator actions for responding to plant alarms.

B. Operating Procedure Program

The SAR or other submittal should describe the applicant's program for developing the operating procedures (A.1-5 above). The staff will review the applicant's program for development and implementation of the operating procedures.

C. Emergency Operating Procedure Program

The SAR or other submittal (e.g., the procedures generation package [PGP]) should describe the applicant's program for developing emergency operating procedures (A.4 above) as well as the required content of the EOPs. The staff will review the applicant's program for development and implementation of the EOPs.

The procedure development program, as described in the PGP for EOPs, should be submitted to the NRC at least 3 months prior to the date the applicant plans to begin formal operator training on the EOPs. The PGP should include:

1. Plant-specific technical guidelines (P-STGs), which are guidelines based on analysis of transients and accidents that are specific to the applicant's plant design and operating philosophy. The submitted documentation of the P-STGs will provide the basis for, and include a reference to, generic guidelines, if used.

For plants not referencing generic guidelines, this section of the submittal should contain the action steps necessary to mitigate transients and accidents in a sequence that allows mitigation without first having diagnosed the specific event, along with all supporting analyses, to meet the requirements of TMI Action Plan Item I.C.1 (NUREG-0737 and Supplement 1 to NUREG-0737).

For plants referencing generic guidelines, the submitted documentation should include (1) a description of the process used to develop plant-specific guidelines from the generic guidelines, (2) identification of significant deviations from the generic guidelines (including identification of additional equipment beyond that identified in the generic guidelines), along with all necessary engineering evaluations or analyses to support the adequacy of each deviation, and (3) a description of the process used for identifying operator information and control requirements. Examples of significant safety deviations are provided in Subsection 3.3.2 to Appendix A to this Standard Review Plan (SRP) section.

2. A plant-specific writer's guide (P-SWG) that details the specific methods to be used by the applicant in preparing EOPs based on P-STGs.

3. A description of the program for verification and validation (V&V) of EOPs.

4. A description of the program for training operators on EOPs.

D. Review Interfaces

IEHB coordinates evaluations by other branches that involve the review of operating procedures as defined in A, above. If an applicant references or provides unreviewed technical guidelines as the basis for the plant-specific EOPs, IEHB will conduct an initial review of the guidelines. Assistance from other technical review branches will be obtained as necessary to perform a thorough review of the safety-significant deviations.

If unapproved guidelines incorporate significant technical changes from approved guidelines, SRXB may request technical review by the SPLB. SRXB and SPLB will develop requests for additional information, if necessary, and will provide safety evaluation (SE) input to IEHB.

Paperwork Reduction Act Statemement

The information collections contained in this NUREG are covered by the requirements of 10 CFR Parts 50 and 52 which were approved by the Office of Management and Budget, approval numbers 3150-0011 and 0151.

Public Protection Notification

If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

II. ACCEPTANCE CRITERIA

Section 13.5.2.1 of the SAR provides additional evidence of the applicant's technical qualifications, and forms a basis for a key part of the regulatory inspection program. Acceptance is based on meeting the relevant requirements of 10 CFR 50.34 as indicated below. Additional guidelines listed in this subsection provide guidance to applicants for meeting basic requirements.

A. Operating Procedure Schedule

A generally acceptable target date for completion of operating procedures is about 6 months before fuel loading to allow adequate time for plant staff familiarization and to allow NRC staff adequate time to develop operator license examinations. The PGP for EOPs must be submitted not later than 3 months prior to the date formal operator training on EOPs is to begin.

B. Control Room and Plant Procedures

The regulations and staff guidelines applicable to operating procedures to be used in the control room and locally in the plant are as follows:

1. 10 CFR 50.34(a)(6) and (10) and 10 CFR 50.34(b)(6)(iv) and (v).

2. 10 CFR Part 50, Appendix B, Criteria V and VI, establish criteria for development, approval, and control of procedures for all activities affecting quality.

3. The review criteria for procedures in NUREG-0711, Chapter 9, "Element 8 - Procedure Development."

4. NUREG-0737, "Clarification of TMI Action Plan," Item I.C.1, "Guidance for the Evaluation and Development of Procedures for Transients and Accidents." (Emergency Operating Procedures Only)

5. Supplement 1 to NUREG-0737, TMI Action Plan Items I.C.1 and I.C.9, "Requirements for Emergency Response Capability," Item 7, Subsections 7.1 and 7.2, "Upgrade of Emergency Operating Procedures." (Emergency Operating Procedures Only)

6. The guidelines in the Regulatory Position section of Regulatory Guide 1.33.

7. The guidelines of ANSI/ANS 3.2-1982, Section 5.3.

8. Appendix A to Standard Review Plan, Section 13.5.2.1, "Guidelines for the Evaluation of Procedures Generation Packages." (Emergency Operating Procedures Only)

9. Supplement 1 to NUREG-1358, "Lessons Learned from the Special Inspection Program for Emergency Operating Procedures," 1992.

C. Technical Rationale

The technical rationale for application of these acceptance criteria to operating procedures is discussed in the following paragraphs:

1. Compliance with the requirements of 10 CFR 50.34(a)(6) and (10) and 10 CFR 50.34(b)(6)(iv) and (v) requires that the applicant include in the SAR preliminary plans for emergency organization, training, conduct of operations, and coping.

   Sections 50.34(a)(6) and (10) and 50.34(b)(6)(iv) and (v) of 10 CFR are applicable to this SRP section because they specify in general terms the information to be submitted in the SAR regarding the operating procedure program, an important part of the safe conduct of operations for emergency and nonemergency activities.

   Meeting these requirements provides assurance that the conduct of operations at the plant will be formalized with procedures covering normal and emergency activities. The planning and implementation of a procedure program will provide means for correct and standardized performance of activities important to safety.

2. Compliance with the requirements of 10 CFR Part 50, Appendix B, Criteria V and VI, requires that activities affecting quality be prescribed by documented instructions, procedures, and drawings and that measures be established to control issuance of and changes to these documents.

   Criteria V and VI are applicable to this section because they require an applicant to ensure that quality assurance considerations are an integral part of the operating procedure program governing the development of technical procedures, V&V, implementation, and document control relative to the safe operation of the facility under routine, off-normal, and emergency operating conditions.

   Meeting these requirements provides assurance that activities affecting quality will be satisfactorily controlled.

III. REVIEW PROCEDURES

Review of the SAR or other submittal in accordance with this section consists of a detailed comparison of the information submitted with the acceptance criteria of Subsection II above. The SAR review should encompass only the schedules for procedure development and determination that the applicant commits to follow the applicable regulatory guides and standards.

(The following paragraph is applicable to all operating procedures as described in Section I.A above)

Review the applicant's program for the development of operating procedures to ensure the application of accepted human factors principles and practices for the design of the operating procedures. Element 8 of NUREG-0711, "Procedure Development", describes an acceptable method for developing operating procedures which is an integral part of the human factors engineering (HFE) program. The HFE program is described more fully in Chapter 18 of the SRP.

(The following paragraph is applicable to EOPs only)

To supplement the expertise of the reviewer, especially in the human factors area, and to promote consistency among the PGP reviews, Appendix A identifies the subjects which should be considered by the reviewer in the evaluation. However, Appendix A is not a "checklist" and an acceptable PGP need not be address each item of Appendix A.

Normally the PGP review should be conducted prior to the date the applicant plans to begin formal operator training on the EOPs. If this is not possible because of a delayed submittal, perform an acceptance review of the PGP. Specifically, audit the four parts of the PGP to determine if there are any major deficiencies in the EOP program that warrant postponing operator training. If major deficiencies are found, identify the additional information necessary to conduct the complete PGP review to the Licensing Project Manager so that the applicant can be notified prior to the initiation of training.

Review the PGPs to determine if the applicant's program meets the requirements of Generic Letter 82-33. The review consists of the evaluation of the four parts of the PGP: the P-STGs, the P-SWG, the description of the program for V&V, and the description of the training program necessary to support the conclusions described in Subsection IV below. To support this review, Appendix A provides additional review guidance.

Review the P-STGs to determine if acceptable analyses of accidents and transients and development of technical guidelines for operator actions applicable to the plant have been completed, and to determine if an acceptable process for identifying operator information and control needs has been described. The Human Factors Engineering Program Review Model (HFE PRM), as described in NUREG-0711, provides additional guidance on review of applicant procedure development programs. It is expected that most applicants will reference generic technical guidelines.

For an applicant using approved generic technical guidelines as the basis for its P-STGs, the major portion of the review of the technical guidelines has been accomplished generically. Staff SERs approving for use each of the four owners groups' generic technical guidelines have been published and may be supplemented as guidelines are revised. The review of this type of P-STGs should focus on the process described for converting generic technical guidelines into plant-specific procedures to ensure that the safety-significant deviations from the generic guidelines are controlled. The evaluation should include the technical adequacy of the identified plant-specific deviations. Finally, the process should be evaluated for development of the plant-specific information and control requirements necessary to use the EOPs.

The review of identified safety-significant deviations from generic technical guidelines will be conducted to the same level of detail as the generic technical guidelines. Examples of safety-significant deviations are given in Appendix A, Subsection 3.3.2. Assistance from other technical review branches will be obtained as necessary to perform a thorough review of the safety-significant deviations. Only safety-significant deviations need to be reviewed. However, the reviewer will determine that the applicant's program will control this process so that the work is auditable. It is expected that most applicants will control the process by documenting all deviations.

Since B&W plant owners elected to use a lead plant concept rather than generic technical guidelines, each B&W applicant's identified deviations from the lead plant's (Oconee's) guidelines will be reviewed.

For applicants not referencing generic technical guidelines, ensure that the submittal includes analysis of accidents and transients in accordance with the guidance of NUREG-0660, 10 CFR 50.34(f)(2)(ii), and NUREG-0737, Items I.C.1 and I.C.9. To do this, (1) become familiar with the integrated performance of the NSSS and balance-of-plant systems, (2) evaluate the completeness of the accident and transient analyses, (3) evaluate the use of appropriate models, calculational methods, and plant data, (4) consider audit calculations of selected accidents and transients (assistance from other technical review branches required), (5) evaluate the adequacy of the applicant's program to develop guidelines from the analysis of accidents and transients, (6) test the guidelines against scenarios, including multiple failures, and (7) evaluate the information and control needs of the operators to execute the instructions of the guidelines. NUREG-0711 provides guidance on analyses appropriate for human-system interaction requirements. (Refer to Chapter 18 for additional information.)

The P-SWG review will consider the adequacy of the methods of presentation of the technical information in the EOPs to ensure that the EOPs are complete, accurate, consistent, and easy to understand and follow for the intended users (e.g., control room operators, shift supervisors, and auxiliary operators). Review the P-SWGs by evaluating the applicant's methods for meeting the overall writer's guide objectives stated in NUREG-0899 and the objectives of NUREG-0711, Chapter 9, "Procedure Development," and criteria described in Appendix B of NUREG-1358, Supplement 1. Appendix A provides guidance to assist the reviewer in making this evaluation. This guidance is not to be used as a set of strict criteria, but is to be used as an aid in the overall evaluation of the P-SWG. Because strict criteria do not exist for the human factors evaluation, the reviewer must make a professional judgment regarding the adequacy of the applicant's methods as described in the P-SWGs.

Review the V&V and training programs by comparing the program descriptions with the objectives of NUREG-0899 and NUREG-0711.

The level of effort for these PGP reviews will vary significantly. For example, the effort necessary to review the P-STGs will vary depending on the number, complexity, and significance of the plant-specific deviations from the approved generic technical guidelines.

If the review of the PGP does not yield sufficient information to support the conclusions of the Evaluation Findings section, the reviewer should obtain at least one EOP for review. As a product of the PGP program, the EOP or EOPs would then be additional information for judging the program's acceptability and will provide additional information as to how the applicant's EOP development and implementation program should be modified to ensure that it contains sufficient information to assure acceptability of the resulting EOPs.

When the reviewer has determined that each of the criteria of Subsection II has been satisfied based upon the statements made by the applicant in the SAR, the review of Section 13.5.2.1 is complete.

For standard design certification reviews under 10 CFR Part 52, the procedures above should be followed, as modified by the procedures in SRP Section 14.3, to verify that the design set forth in the standard safety analysis report, including inspections, tests, analysis, and acceptance criteria (ITAAC), site interface requirements, and combined license action items, meets the acceptance criteria given in Subsection II. SRP Section 14.3 (proposed) contains procedures for the review of certified design material (CDM) for the standard design, including the site parameters, interface criteria, and ITAAC.

IV. EVALUATION FINDINGS

The reviewer verifies that the information presented and the review support the following type of conclusion, to be used in the staff's safety evaluation report:

> The applicant's program for operating procedures as described in the SAR is in accordance with 10 CFR 50.34, Regulatory Guide 1.33, and ANSI/ANS 3.2-1982, Section 5.3, and is acceptable. The staff reviewed the applicant's program for development of operating procedures and reached the following conclusions:
>
> 1. With respect to technical guidelines:
>
>   (a) The operating procedures will be based upon acceptable technical-guidance-derived plant design bases, system-based technical requirements and specifications, task analysis results, and critical human actions identified in the HRA/PRA.
>
>   (b) The EOPs will be based upon acceptable technical guidelines derived from approved analyses of transients and accidents.
>
>   (c) Implementation of the applicant's described methods for conducting an analysis of the operator's tasks should result in the identification of the instrumentation and controls necessary to perform the tasks specified in the technical guidelines.
>
> 2. With respect to writer's guidance:
>
>   (a) The writer's guide or guides provides sufficient information to help ensure that operating procedures, including EOPs, developed using technical guidelines will be complete, accurate, consistent, and easy to understand and follow.
>
>   (b) The methods described by the writer's guide appear sufficient to support upgrading of the operating procedures, including EOPs, and to ensure long-term consistency within and among these procedures.
>
> 3. Implementation of the described V&V program provides adequate assurance that the operating procedures, including EOPs, are technically correct and useable, follow the applicable writer's guide correspond to the control room/plant hardware, and are compatible with the minimum number, qualifications, training, and experience of the operating staff.
>
> 4. Implementation of the described training program should result in the operator understanding the philosophy behind the approach to the operating procedures, including EOPs, understanding the mitigative strategy of the EOPs and technical

basis of the operating procedures, having a working knowledge of the technical content of the operating procedures, including EOPs, and having the capability to execute the operating procedures, including EOPs, under operational conditions.

The evaluation findings for this section should also include the following:

1. A statement that the applicant has committed to operate the plant in accordance with written and approved procedures.

2. A brief description of the categories of procedures to be included.

3. A description of the review conducted to ensure that to NUREG-0737, Supplement 1, Item 7, "Upgrade of Emergency Operating Procedures," has been implemented.

For design certification reviews, the findings will also summarize, to the extent that the review is not discussed in other safety evaluation report sections, the staff's evaluation of inspections, tests, analyses, and acceptance criteria (ITAAC), including design acceptance criteria (DAC), site interface requirements, and combined license action items that are relevant to this SRP section.

V. IMPLEMENTATION

The following is intended to provide guidance to applicants and licensees regarding the NRC staff's plans for using this SRP section.

This SRP section will be used by the staff when performing safety evaluations of license applications submitted pursuant to 10 CFR Part 50 or 10 CFR Part 52 and applications for modifications to systems or functions pursuant to 10 CFR 50.59. Except when the applicant proposes an acceptable alternative method for complying with specified portions of the Commissions' regulations, the method described herein will be used by the staff in its evaluation of conformance with the Commission regulations.

The provisions of this SRP section apply to reviews of applications docketed 6 months or more after the date of issuance of this SRP section.

Implementation schedules for conformance to parts of the methods discussed herein are contained in the referenced regulatory guidesand NUREGS.

The staff will use this SRP for judging the acceptability of an applicant's operating procedure program, including the EOP [PGP] program, as described in submittals made in accordance with Supplement 1, NUREG-0737, “Requirements for Emergency Response Capability” (Generic Letter 82-33). The review guidance in this SRP section replaces the review guidance in Generic Letter 82-33.

It is recognized that development of detailed procedures and associated training materials may be beyond the scope of design certification and therefore would be the responsibility of an applicant referencing the certified design.

VI. REFERENCES

1. 10 CFR 50.34, “Contents of Applications; Technical Information.”

2. 10 CFR Part 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants."

3. NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," 1980.

4. NUREG-0711, "Human Factors Engineering Program Review Model," 2002.

5. NUREG-0737, "Clarification of TMI Action Plan Requirements," 1980.

6. NUREG-0737, Supplement 1, "Requirements for Emergency Response Capability," 1983 (Generic Letter 82-33, December, 1982).

7. NUREG-0899, "Guidelines for Preparation of Emergency Operating Procedures," 1982.

8. Generic Letters 83-05, 83-22, 83-23, and 83-31, Staff Safety Evaluation Reports for Generic Technical Guidelines for GE, CE, W, and B&W plants, respectively.

9. Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)."

10. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants."

11. ANSI/ANS 3.2 1982, "Standard for Administrative Controls for Nuclear Power Plants," American National Standards Institute.

12. NUREG-1358, "Lessons Learned From the Special Inspection Program for Emergency Operating Procedures," 1989.

13. NUREG-1358, Supplement 1, "Lessons Learned From the Special Inspection Program for Emergency Operating Procedures," 1992.

Appendix A to SRP Section 13.5.2.1

REVIEW PROCEDURES FOR THE EVALUATION
OF PROCEDURES GENERATION PACKAGES

1.0 Background

In August of 1982, NUREG-0899, "Guidelines for the Preparation of Emergency Operating Procedures," was published. This document is designed to "identify the elements necessary for licensees and applicants to prepare and implement Emergency Operating Procedures (EOPs) that will provide the operator with directions to mitigate the consequences of a broad range of accidents and multiple equipment failures." In addition to identifying these elements, the document also outlines the process by which licensees and applicants should develop, implement, and maintain EOPs. To ensure that the elements are addressed in the new or upgraded procedures and that acceptable processes of development, implementation, and maintenance are used, the staff identified a method of review that is intended to provide confidence that EOPs written or upgraded according to a given plant's program would be acceptable. The NRC staff believes that it is more important that licensees and applicants ensure that the process used to generate procedures and the technical basis for the procedures are sound and well documented, than to perform a one-time review of EOPs, with no assurance that future EOP revisions will be technically adequate and consistent with existing EOPs. With this approach, responsibility for the generation and review of the EOPs, as well as future revisions to EOPs, is retained by the licensee.

In NUREG-0899, four aspects of EOP development and implementation are identified as providing an adequate basis for review. These are (1) plant-specific technical guidelines (P-STGs); (2) a plant-specific writer's guide; (3) a description of the program for verification and validation of the EOPs; and (4) a description of the program for training operators on the EOPs. Information on each of these items is to be provided in the procedures generation package (PGP). The PGP for each plant will provide the licensee with a technical and human factors basis for developing its EOPs and for making future revisions to its EOPs.

The formal requirement for submitting this package is provided in Supplement 1 to NUREG-0737, "Requirements for Emergency Response Capability" (Generic Letter No. 82-33).

In 1994, NUREG-0711, "Human Factors Engineering Program Review Model" (HFE PRM), was published. The HFE PRM, described more fully in SRP Chapter 18, contains guidance on reviewing human factors engineering program elements, including procedure development (Chapter 9). The HFE PRM addresses technical procedures, including abnormal and emergency procedures, and seeks to ensure that an "applicant's procedure program will result in procedures that support and guide human interaction with plant systems and control plant-related events and activities." Therefore it is important that human-system interaction issues be considered in the development of all procedures, including all operating procedures (described in I.A of SRP Section 13.5.2.1) to be used within the control room and locally in the plant, including emergency operating procedures (EOPs).

The guidance contained here in SRP Section 13.5.2.1, Appendix A, specifically addresses EOPs. Emergency operating procedures are particularly important for safety in nuclear power plant operation. However, it should be recognized that all technical procedures need to be developed to assist personnel in performing tasks. Elements to consider more broadly can be found in NUREG-0711. Other documents that may be used as guidance in the review of procedures include those referenced in the References section of this appendix.

The purpose of this document is to provide guidance for reviewers during their evaluation of PGPs. The PGP is expected to contain specific information in each of its four parts. The review guidance below is divided into general objectives and specific review guidelines. The listing of review guidelines represents what the staff believes should be considered by reviewers in determining if the general objectives are met. Because each of the objectives can be adequately addressed in many ways and may be satisfied without addressing each of the review guidelines, it will often be necessary for reviewers to use their expert judgment in determining the acceptability of a particular submittal. The general objectives and supporting documents such as NUREG-0899 and NUREG-1358, Supplement 1, should be used as guidance in making these judgments. The methods provided in NUREG-0899 and in Appendix B to NUREG-1358, Supplement 1, are an acceptable approach for preparing EOPs. It should be recognized, however, that approaches other than those found in these documents may be acceptable, and reviewers will need to use their judgment in determining the adequacy of the PGP.

As described in the SRP, all PGPs will be reviewed by the staff. The review guidelines presented in Subsections 3 through 6 of this appendix provide additional assistance to the reviewers. All applicants have the option of providing a justification for their approach where they disagree with a staff position. When all issues are resolved or when the schedule dictates, the reviewer will prepare a safety evaluation report (SER).

2.0 General Guidance to Reviewers

The guidance that follows is provided to assist the reviewer in using the criteria presented in Subsections 3 through 6 of this appendix.

2.1 Reviewers should be aware that different degrees of objectivity (and thus, subjectivity) may be required in reviewing each of the four parts of the PGP since the parts may differ in detail and approach.

2.2 Reviewers should become very familiar with the general objectives associated with each section of a PGP. The specific review guidelines can serve as the basis for making the subjective evaluations of the general objectives.

2.3 When an objective is not met or a specific response cannot be judged acceptable because of missing information, the reviewer should identify the information that is missing and what is needed to make the PGP acceptable.

2.4 Some items included in a PGP may not be addressed within either the general objectives or the specific review guidelines. These items must be evaluated carefully to ensure that unnecessary or possibly detrimental inclusions do not occur in the EOPs (e.g., an EOP Deficiencies section is not a desirable inclusion in an EOP).

2.5 As stated in the Background section, most of the review guidelines are subjective in nature. The reviewer will have to judge whether the discussion of an item is sufficiently clear, complete, and technically acceptable to achieve the objectives.

2.6 In some instances the language (i.e., names, titles, etc.) used in the PGPs may be different from that used in this document, although the same subjects or items are being discussed. For example, the format of "decision aids" may be covered under a PGP section with the heading, "Job Performance Aids." Reviewers should be careful that identified PGP deficiencies are not based on semantics.

2.7 In some instances a particular subject may appear not to be addressed in the PGP, when in fact it is addressed in another part of the PGP. For example, the determination of the adequacy of control room instrumentation and controls may not be addressed in the P-STGs, but included as a part of the validation and verification program. Reviewers must therefore become familiar with the general objectives and specific review guidelines as a whole so that these situations can be readily identified.

3.0 Plant-Specific Technical Guidelines

3.1 General Discussion

All licensees and applicants are required to submit P-STGs. These guidelines may be based on (1) generic technical guidelines (prepared by the owners group), or (2) a plant-specific reanalysis of transients and accidents as described in TMI Action Plan Item I.C.1. In either case, the P-STGs should be based on the identification of plant systems and functions, and be supported by an analysis of operator tasks to identify operator information and control needs. Among the four approved generic technical guidelines, operator task information is provided using different levels of detail. If generic technical guidelines are referenced, the need for additional task specification will be different depending upon the level of task information provided by the generic technical guidelines and the nature of deviations from the guidelines.

The information to be submitted in the PGP as P-STGs is dependent on whether or not generic technical guidelines are used, as well as the degree to which plant-specific characteristics (e.g., equipment) are consistent with the plant on which the generic technical guidance is based.

Some of the "deviations" that must be addressed as part of the P-STG submittal are differences between the generic technical guidelines and the P-STGs. This includes differences due to plant initiatives and those identified in the generic guidelines as "plant-specific" items. Only differences that are safety significant, e.g., related to systems functions, or methods, should be reviewed. Subsection 3.3.2 provides examples of other deviations that must also be addressed. Where an applicant does reference NRC-approved generic technical guidelines, the applicant should not submit those guidelines. However, safety-significant deviations from the mitigative strategy should be described. Furthermore, applicants using generic guidelines need not submit the detailed action steps. The process for developing the action steps from the generic guidelines should be described. Applicants not using generic guidelines should submit, as a part of the P-STGs, the action steps necessary to mitigate transients and accidents, and supporting technical analysis and bases. The P-STGs should have an orientation that allows mitigation without event diagnosis. In either case, the applicant should submit a description of how operator information and control needs were derived and used to specify instrumentation and control requirements.

The guidance presented below identifies elements reviewers should consider in determining acceptability of P-STGs.

3.2 General Technical Objectives

The purpose of the review of the technical guidelines submittal is to determine that the following general objectives are adequately addressed. Specific evaluation elements are identified in Subsections 3.3 and 3.4.

3.2.1 The EOPs will be based on acceptable technical guidelines derived from approved analyses of transients and accidents as described in NUREG-0660, Items I.C.1 and I.C.9, as clarified by Item I.C.1 in NUREG-0737 and Supplement 1 to NUREG-0737. The P-STGs, the generic guidelines (if referenced), and supporting documentation provide EOP writers with all the technical information necessary for preparing EOPs which direct operators' actions to mitigate the consequences of transients and accidents without a need to first diagnose an event to maintain the plant in a safe condition (function orientation).

Part of the acceptability of the P-STGs is that the P-STGs are validated by the applicant using methods acceptable to the reviewer (see NUREG-0899, Sections 2.6 and 4.2).

3.2.2 The PGP describes an adequate method to identify information and control needs and to provide a basis for identifying control room instrumentation and controls necessary to perform the tasks specified in the technical guidelines.

3.3 Specific Review Guidelines - Plants Using NRC-Approved Generic Technical Guidelines

To determine that the applicant's PGP adequately accomplishes the above objectives, the reviewer should consider the following:

3.3.1 P-STG development

3.3.1.1 Approved version of generic technical guidelines indicated

3.3.1.2 A description of the process used to translate the generic technical guidelines into the P-STGs

3.3.2 Deviations and additions

3.3.2.1 Identification of safety-significant deviations from the NRC-approved generic technical guidelines. The following are examples of deviations that should be considered:

a. any modification to the mitigative strategy of the generic technical guidelines (e.g., for a Westinghouse plant, depressurizing the RCS following a steam generator tube rupture without first having conducted a limited cooldown in accordance with the guidelines to establish a margin to saturation)

b. differences in equipment operating criteria (e.g., RCP trip criteria, SI injection termination criteria)

c. differences in equipment operating characteristics (i.e., between the plant-specific equipment and that assumed in the generic analyses, such as SI that can be throttled vs. only on/off)

d. identification of methods and equipment used to address the technical areas of the generic guidelines that are specified as "plant-specific"

e. plant-specific setpoints or action levels that are calculated or determined in a manner other than specified in the generic technical guidelines

NOTE: Plant-specific setpoints (e.g., setpoints associated with automatic initiation of ECCS) called for by the generic guidelines need not be included in the P-STG submittal.

f. actions that are taken in addition to those specified in the generic guidelines and that affect the mitigative strategy

1. differences that affect the equipment's ability to adequately provide the necessary mitigative function

2. use of different instruments or control parameters than those specified in the generic technical guidelines or determination of instrumentation and control characteristics in a manner different than, or with a different basis than, that specified in the generic technical guidelines

3.3.2.2 Identification of items not covered by the NRC-approved generic technical guidelines (e.g., plant-specific conditions, equipment, operations, or [bracketed] information from the generic technical guidelines that relate to systems, functions, or methods)

3.3.2.3 Indication that the safety-significant deviations and additions have been identified and technically justified

NOTE: The reviewer has the option of either reviewing the complete P-STGs with associated technical justification or reviewing only the identified deviations from generic technical guidelines, including technical justification consistent with the Generic Letter 82-33 requirements.

3.3.3 Technical adequacy of operator actions (not covered by, or deviations from, the generic technical guidelines)

NOTE: The evaluation of the technical adequacy of operator actions (i.e., that the procedures will work) may be addressed in the validation and verification sections of the PGP (i.e., at the completion of EOP development rather than during EOP development). The P-STG portion of the PGP should describe how the licensee will determine if the approach taken is effective in mitigating transients and accidents.

3.3.3.1 Description of the verification and validation of operator actions (to determine their technical adequacy)

3.3.4 Applicant's determination of the need for and the adequacy of control room instrumentation and controls for emergency operations

3.3.4.1 Description of the method used to determine information and control needs of the operators (function and task analysis)

NOTE: The determination of the adequacy of control instrumentation and controls may be addressed in the validation and verification sections of the PGP (i.e., at the conclusion of EOP development rather than during EOP development). For the P-STGs, adequacy of control room instrumentation and controls means that the available instrumentation and controls have been evaluated against the information and control needs of the operators and it has been determined that the parameters are correct and that the instrument and control characteristics (e.g., instrument range, units, precision, rate, and setpoints; control type, function, rate, gain, and response) meet the needs identified.

3.3.4.2 Description of the method used to determine if the control room instrumentation and controls meet the information and control needs of the operators

3.4 Specific Review Guidelines - Plants Not Using Generic Guidelines

The review of the P-STGs for plants not referencing generic guidelines will be performed using a methodology similar to that used to evaluate the acceptability of the owners group guidelines. The reviewer should evaluate analyses submitted to support proposed accident recovery strategies, including any analytical models. Improvements in accident recovery techniques should be encouraged; however, in the review of alternate strategies, the reviewer should obtain from the applicant sufficient technical bases to demonstrate that the plant remains within its SAR licensing basis envelope (for licensing basis events).

The reviewer evaluates the effects of, and resulting recovery strategies, for transients and accidents, using the guidance available in NUREG-0737. The P-STG reviewer should consider the following:

3.4.1 Analysis of transients and accidents (consistent with requirements of NUREG-0660 and NUREG-0737)

NOTE: The steps to be taken for this review are contained in the Review Procedures, SRP Section 13.5.2.1.

3.4.2 Validation of technical adequacy of operator actions

NOTE: The evaluation of the technical adequacy of operator actions (i.e., that the procedures will work) may be addressed in the validation and verification sections of the PGP (i.e., at the completion of EOP development rather than after P-STG development). The P-STG portion of the PCP should describe how the applicant will determine if the approach taken is effective in mitigating transients and accidents.

3.4.2.1 Description of the validation or verification of operator actions

3.4.3 Determination of the need for and the adequacy of control room instrumentation and controls for emergency operation

3.4.3.1 Description of the method used to determine information and control needs of the operators

NOTE: The determination of the adequacy of control room instrumentation and controls may be addressed in the validation and verification sections of the PGP (i.e., at the conclusion of EOP development rather than after P-STG development) or in the part of the SAR addressing the human factors engineering of plant systems (SRP Chapter 18). For the P-STGs, adequacy of control room instrumentation and controls means that the available instrumentation and controls have been evaluated against the information and control needs of the operators and it has been determined that the parameters are correct and that the instrument and control characteristics (e.g., instrument range, units, precision, rate, and setpoints; control type, function, rate, gain, and response) meet the needs identified.

3.4.3.2 Description of the method used to determine if the control room instrumentation and controls meet the information and control needs of the operators.

4.0 Review of the Plant-Specific Writer's Guide

4.1 General Discussion

Applicants are required to submit a writer's guide that details the specific methods to be used in preparing EOPs which are based on the P-STGs. NUREG-0899 provides the objectives and purpose of the writer's guide. Appendix B of NUREG-1358, Supplement 1, provides additional criteria useful in developing a writer's guide. Because of the variety of available technical writing style guides and other references pertaining to the presentation of information, the specific information found in the writer's guide is expected to vary considerably among plants. To supplement the human factors expertise of the reviewer, review guidelines are provided that address instructions and guidance expected to be found in writer's guides. In addition, the writer's guide should contain general, philosophical standards and information which would assist the writers in preparing the EOPs.

4.2 General Writer's Guide Objectives

The purpose of the evaluation is to determine if acceptable methods are described for accomplishing the following general objectives.

4.2.1 The writer's guide provides sufficient information for using the P-STGs to develop EOPs , which are useable, accurate, complete, readable, convenient to use, and acceptable to control room personnel.

4.2.2 The writer's guide supports upgrading of the procedures and long-term consistency within and between procedures.

4.3 Specific Review Guidelines

The number in parentheses following each element designates the specific section within NUREG-0899 where the element is addressed.The items with asterisks may appear in a procedure at the discretion of the applicant. If they are used in the EOPs, they should be addressed in the writer's guide and considered in the review. Where a sample procedure is submitted as a part of the writer's guide, the reviewer should verify that any nonrequired element included in the procedure is addressed in the writer's guide.

To determine that the applicant's PGP includes methods which appear adequate to accomplish the above objectives, the reviewer should consider the following:

4.3.1 Organization, content, and format of major sections of the EOPs (5.5)

4.3.1.1 Cover page (5.4.1)

4.3.1.1 Table of contents* (5.4.2)

4.3.1.3 Scope statement (5.4.3)

4.3.1.4 Entry conditions (5.4.4)

4.3.1.5 Automatic actions* (5.4.5)

4.3.1.6 Content and format of operator action steps, including (a) simple action steps, (b) steps which verify an action, (c) steps of continuous or periodic concern/applicability, (d) steps for which a number of alternative actions are equally acceptable, (e) steps performed concurrently with other steps, and (f) steps which lead the operator to the appropriate subsection of the EOPs (5.4.6, 5.4.7, 5.7, 5.8)

4.3.1.7 Figures and tables* (5.4.8 and 5.5.8)

4.3.1.8 Flowcharts and decision aids* (5.4.8 and 5.5.9)

4.3.1.9 EOP page identifying information, including title, procedure number, revision number and date, number of pages, unit designation (if applicable), facility designation, and location of identifying information in the EOP (5.5.1)

4.3.1.10 Page layout, including margins, line spacing, and steps complete on page (5.5.2)

4.3.1.11 Warnings (or cautions) and notes, including placement, definitions, emphasis and format, and complete on one page (5.3, 5.7.9, 5.7.10)

4.3.1.12 Placekeeping aids (5.5.4)

4.3.1.13 Emphasis techniques (5.5.6)

4.3.1.14 Divisions, headings and numbering of pages and steps (5.5.5)

4.3.2 Writing Style (5.6)

4.3.2.1 A vocabulary list - words to use, with definitions, and words to avoid (5.6.1)

4.3.2.2 A list of abbreviations, acronyms, and symbols, and label consistency between procedures and control room (5.6.2)

4.3.2.3 Sentence structure and limit on actions per step (5.6.3)

4.3.2.4 Punctuation (5.6.4)

4.3.2.5 Capitalization (5.6.5)

4.3.2.6 Units of measure in the action steps and in the tables and figures should be consistent with presentation of information in the control room (5.6.6).

4.3.2.7 Numerals, including type, use of decimals and significant digits (5.6.7)

4.3.2.8 Tolerances (5.6.8)

4.3.2.9 Formulas and calculations* (5.6.9)

4.3.2.10 Titles/nomenclature of instrumentation and controls (what information to provide in the procedure and in what format) (5.6.2)

4.3.3 Conditional and logic statements, including format, style, emphasis; definition and use of logic terms; and logic terms and sequences to avoid (5.6.10 and Appendix B)

4.3.4 Referencing other procedures, sections of procedures or subprocedures, and specific steps of procedures (5.2.2 and 5.5.7)

4.3.4.1 Content and format of reference (5.2.2)

4.3.4.2 The criteria used to determine when steps of a referenced procedure are to be included in an EOP (to minimize cross-referencing) (5.2.2).

4.3.4.3 Method for identifying sections or subsections (e.g., use of tabbing) (5.5.7 and 6.1.4)

4.3.5 When and how to present location Information (equipment, controls and displays) (5.7.11)

4.3.6 Control Room Staffing and Division of Responsibilities (5.8)

NOTE:This section addresses the need to consider operating crew staffing and responsibilities during the process of developing EOPs to help ensure efficient and effective implementation of EOPs during an emergency. Deficiencies in this regard may be identified by the applicant during validation or verification of the EOPs. Subsection items 4.3.6.1 through 4.3.6.4 may therefore be addressed under validation and verification.

4.3.6.1 Structuring of EOPs to ensure that minimum staffing can execute the EOPs

4.3.6.2 Designating the operators' responsibilities in implementing EOPs (i.e., each operator will know what he or she has to do during an emergency; it is not necessary to specify roles in PGP or EOPs)

4.3.6.3 Sequencing action steps to minimize physical interference between operators

4.3.6.4 Sequencing action steps to avoid their unintentional duplication by operators

4.3.7 Use and maintenance of EOPs, including accessibility and quality of copies (6.0)

4.3.8 Statement of commitment to use writer's guide in developing and revising the EOPs

5.0 Program for Validation and Verification

5.1 General Discussion

All applicants must submit a description of their programs for validating and verifying their EOPs. NUREG-0711, Element 10, Human Factors Verification and Validation, provides additional guidance on the development of a verification and validation program. Both technical and human factors aspects of the EOPs are addressed by validation and verification activities, and submittals may integrate the two aspects under a given evaluation scheme. For these reasons reviewers will have to exercise considerable judgment in their review of the submittals. The evaluation elements for validation and verification were drawn from the six objectives identified in NUREG-0899 (Subsection 3.3.5.1). These objectives, which are repeated below, should serve as the general basis for determining the acceptability of the validation and verification programs reviewed.

5.2 General Objectives

The purpose of evaluating the validation and verification program is to ensure that the following general objectives are met. A listing of specific evaluation elements is provided in Subsection 5.3.

5.2.1 EOPs are technically correct, i.e., they accurately reflect the technical guidelines.

5.2.2 EOPs are written correctly, i.e., they accurately reflect the plant-specific writer's guide.

5.2.3 EOPs are useable, i.e, they can be understood and followed without confusion, delays, errors, etc.

5.2.4 There is a correspondence between the procedures and the control room/plant hardware, i.e., controls, equipment, and indications that are referenced are available (inside and outside of the control room), use the same designations, use the same units of measurement, and operate as specified in the procedures.

5.2.5 The language and level of information in the EOPs are compatible with the minimum number, qualifications, training, and experience of the operating staff.

5.2.6 There is a high level of assurance that the procedures will work, i.e., the procedures guide the operator in mitigating transients and accidents

5.3 Specific Validation and Verification Review Guidelines

To aid the reviewer in the evaluation of the validation and verification program, the reviewer should consider the following review guidelines:

5.3.1 The applicant should Indicate the methods that will be used to meet each of the objectives (as specified in Subsection 5.2 above) of the validation and verification program; the specific combination of methods for meeting each objective should be identified by the applicant so that the reviewer has assurance that the objectives of the overall validation and verification program are met. In the staff's judgment, the following combination of methods should be used to meet each of the objectives:

5.3.1.1 Whether the EOPs are technically correct (i.e., whether they accurately reflect the technical guidelines) should be evaluated by a combination of the following methods: (a) desk-top review, and (b) seminars, workshops, operating team review, and computer modeling/analysis.

5.3.1.2 Whether the EOPs are written correctly (i.e., whether they accurately reflect the [approved] plant-specific writer's guide) should be evaluated by a combination of the following methods: (a) desk-top review, and (b) seminars, workshops, and operating team review.

5.3.1.3 Whether there is a correspondence between the procedures and the control room/plant hardware (i.e., controls, equipment, and indications that are referenced are available inside and outside the control room, use the same designations, and the same units of measurement, and operate as specified in the procedures) should be evaluated by a combination of the following methods: (a) seminars, workshops, and operating team review, (b) control room walkthroughs (static), and (c) simulation (if plant-specific) (static).

5.3.1.4 Whether the EOPs are usable (i.e., they can be understood and followed without confusion, delays, errors, etc.) for the given level of qualifications, training, and experience of the control room staff, should be evaluated by a combination of the following methods: (a) seminars, workshops, and operating team review, (b) simulator exercises, and (c) control room walkthroughs (dynamic).

5.3.1.5 Whether the language and level of information presented in the EOPs are compatible with the minimum control room staffing and the qualifications, training, and experience of the control room staff should be evaluated by a combination of the following methods: (a) desk-top review, (b) seminars, workshops, and operating team review, (c) simulator exercises, and (d) control room walkthroughs (dynamic).

5.3.1.6 Whether there is a high level of assurance that the procedures will work (i.e., the procedures guide the operator in mitigating transients and accidents) should be evaluated by a combination of the following methods: (a) desk-top review, (b) seminars, workshops, and operating team review, (c) simulator exercises, and (d) control room walkthroughs (dynamic).

5.3.2 Indication that plant operators, subject matter experts, and procedure writers are involved

5.3.3 Identification of the roles played by the participants (i.e., how operators, subject matter experts, etc., will participate in the validation or verification process) (roles should be based on the specific validation or verification objective being addressed)

5.3.4 Use of scenarios

Indication that the full complement of EOPs are exercised, including multiple failures (simultaneous and sequential), and inclusion of criteria for selecting scenarios

NOTE: Where a generic simulator is used, and to some extent, where a plant reference simulator is used, it will not be possible to fully exercise all parts of the EOPs. In these instances, the PGP should describe the method that the licensee will use to ensure that the validation and verification program will cover areas missed in the simulator exercises. The following element is included to address this issue.

5.3.5 Indication that areas not covered by simulator exercises will undergo validation or verification

5.3.6 Description of the plan for correcting and revising EOPs as a result of the validation or verification and for feedback from simulator exercises, control room walkthrough, desk-top reviews, operating team reviews, and operator training to address accuracy, readability, usability, and completeness of the EOPs

5.3.7 Statement of commitment to validate/verify revisions to EOPs, when appropriate, and the conditions under which revisions should be validated/verified

5.3.8 Description of the method by which multiple units will be handled in the validation and verification process to account for unit differences

NOTE: For multiunit sites, the part of the validation and verification process involving control room walkthroughs and use of operators should be carried out for each unit of a multiunit site to the extent that the units differ in terms of instrumentation, controls, equipment (including the availability, design, labeling, or location of equipment), or any other aspect that may impact plant safety.

5.3.9 Indication that the EOPs will be compatible with minimum control room staffing

5.3.10 Description of the plan by which adequacy (in terms of availability, readability and usability) of control room instrumentation and controls will be determined

5.3.11 Description of the plan by which correspondence between EOPs and control room instrumentation and controls will be determined

5.3.12 Where available instrumentation and controls have *not* been evaluated against the information and control needs of the operators as a part of the P-STGs (see Subsections 3.3.4.2 and 3.4.3.2), they should be evaluated as a part of the validation and verification program. The description of the validation and verification program should include the method that will be used to determine the adequacy of control room instrumentation and controls in meeting the information and control needs of the operators (i.e., it has been determined that the parameters are correct and that the instrument and control characteristics [e.g., accuracy, scaling, etc.] meet the needs identified).

NOTE: Since many aspects of validation and verification can be addressed during operator training, it is anticipated that applicants will combine these activities to make more efficient use of simulator time. Where validation or verification is tied to the EOP training program, it is necessary for applicants to distinctly address validation or verification through a formal process which documents results and provides for feeding this information back into the EOP development process. The PGP should describe this process.

NOTE: Where EOPs are partially validated/verified on a generic simulator, licensees should commit to performing the dynamic portion of the validation and verification of the EOPs if a plant reference simulator becomes available.

6.0 Program for Operator Training on EOPs

6.1 General Discussion

Applicants are to submit descriptions of their planned programs for training operators on EOPs. The purpose of reviewing the EOP training program is to ensure that operators will be trained prior to implementation of the EOPs, and that there is a reasonable assurance that the methods to be used in training are adequate. This determination can be made by verifying that the training program meets the general training objectives identified in Subsection 6.2. To determine that these general objectives are met, the reviewer should consider the specific review guidelines of Subsection 6.3 and of NUREG-0711, Element 9, Training Program Development.

6.2 General EOP Training Program Objectives

The purpose of the evaluation is to determine that the following general objectives are adequately addressed in the training program described by considering the following review guidelines. These guidelines are not intended to represent all the necessary components of an adequate training program, but rather to serve as a basis for assuring the staff that the operators have been trained prior to EOP implementation and that they will be capable of using the EOPs.

6.2.1 Trainees should understand the philosophy behind the approach to the EOPs, i.e., their structure and approach to transient and accident mitigation, including control of safety functions, accident evaluation and diagnosis, and the achievement of safe, stable, or shutdown conditions.

6.2.2 Trainees should understand the mitigation strategy and technical bases of the EOPs, i.e., the function and use of plant systems, subsystems, and components in mitigating transients and accidents.

6.2.3 Trainees should have a working knowledge of the technical content of the EOPs, i.e., they must understand and know how to perform each step in all EOPs to achieve EOP objectives.

6.2.4 Trainees should be capable of executing the EOPs as individuals and teams under operational conditions, i.e., they must be able to carry out an EOP successfully during transients and accidents.

6.3 Specific EOP Training Review Guidelines

The reviewer should consider the following specific review guidelines in evaluating the description of the EOP training program:

6.3.1 Inclusion of training objectives consistent with Subsection 6.2 above

6.3.2 Use of simulator exercises

6.3.2.1 Specification of plant-specific or generic simulation

6.3.2.2 Indication that all EOPs will be exercised by all operators

NOTE:Where a generic simulator is used, and to some extent, where a plant reference simulator is used, it will not be possible to fully exercise all parts of the EOPs. In these instances, the PGP should describe the method that the applicant will use to ensure that the validation and verification program will cover areas missed in the simulator exercises. The following element is included to address this issue.

6.3.2.3 A description of the method for training in areas not covered by simulator exercises

6.3.2.4 Indication of planned operator roles and team work

6.3.2.5 Indication of the use of a wide variety of scenarios (i.e., incorporating multiple simultaneous and sequential failures)

6.3.3 Use of Control Room Walkthrough

6.3.3.1 Indication of walkthrough of all EOPs by all operators

6.3.3.2 Indication of planned operator roles and team work

6.3.3.3 Indication of use of a wide variety of scenarios (i.e., incorporating multiple failures, simultaneous and sequential)

6.3.4 Use of lectures, discussion sessions, and seminars

6.4 Indication that operators will be trained prior to implementation of EOPs

6.5 Indication that operators will be evaluated as part of the training program

7.0 References

NUREG-0711, "Human Factors Engineering Program Review Model," 1994.

NUREG-0737, "Clarification of TMI Action Plan Requirements," 1980.

NUREG-0737, Supplement 1, "Requirements for Emergency Response Capability," 1983.

NUREG-0899, "Guidelines for Preparation of Emergency Operating Procedures," 1982.

NUREG-1358, “Lessons Learned From the Special Inspection Program for Emergency Operating Procedures,” 1989.

NUREG-1358, Supplement 1, “Lessons Learned From the Special Inspection Program for Emergency Operating Procedures,” 1989.

Regulatory Guide 1.33, “Quality Assurance Program Requirements (Operation).”

ANSI/ANS 3.2-1982, “Standard for Administrative Controls for Nuclear Power Plants,” American National Standards Institute.

ATTACHMENT 7

SRP Chapter 18.0
"Human Factors Engineering"

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

18.0 HUMAN FACTORS ENGINEERING

REVIEW RESPONSIBILITIES

Primary - Equipment and Human Performance Branch (IEHB)

Secondary - None

I. AREAS OF REVIEW

The Equipment and Human Performance Branch (IEHB) reviews the human factors engineering (HFE) programs of applicants (e.g., for a construction permit [CP]; operating license [OL]; standard design certification [DC]; and combined license [COL]) and licensees (e.g., for modifications and changes to a licensee's design or licensing basis). The purpose of these reviews is to improve safety by verifying that accepted HFE practices and guidelines are incorporated into the plant's design. The guidance provided in this document, and in the supporting documents referenced, is used to conduct these HFE reviews.

This chapter describes a process for evaluating (1) designs, (2) design processes, (3) design reviews, and (4) operator actions submitted by applicants and licensees for the broad range of NRC review responsibilities. Specific applications are discussed in "Applications" below. The chapter identifies 12 areas of review that are needed for successful integration of human characteristics and capabilities into nuclear power plant design. These areas of review include:

- HFE Program Management
- Operating Experience Review
- Functional Requirements Analysis and Function Allocation
- Task Analysis
- Staffing and Qualifications
- Human Reliability Analysis

- Procedure Development
- Training Program Development
- Human-System Interface Design
- Human Factors Verification and Validation
- Design Implementation
- Human Performance Monitoring

While the process defines 12 areas of review, not all may be applicable to reviewing a particular applicant's or licensee's HFE program. This is discussed in "Graded Approach to Review" below.

A. Applications

NRC HFE reviews in three application areas are described below.

1. *Review of the HFE Aspects of a New Plant* - If an applicant proposes to build a new plant under 10 CFR Part 50 requirements, an HFE review of the new license application is performed. This chapter describes the staff's review activities to verify that accepted HFE principles are incorporated during the design process and that the human-system interfaces (HSIs) reflect a state-of-the-art HFE design.

   Nuclear power plant (NPP) designers and vendors may submit designs of advanced standardized NPPs to the NRC for review and approval under 10 CFR Part 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants," (see Part 52 Subpart B, "Standard Design Certification"). To obtain a standard design certification under Part 52, applicants submit a standard safety analysis report (SSAR), which should include information on the HFE program. However, since technology is continually advancing, details of the applicant's HFE design might not be complete before the NRC issues a design certification. In such cases, reviews under 10 CFR Part 52 would primarily focus on the HFE design process.

   An applicant may obtain a COL to operate a standardized NPP that has already received a design certification under 10 CFR Part 52. Aspects of the design not complete at the time of design certification are reviewed at the COL stage. Thus, for advanced NPPs, HFE reviews can occur at different points within the 10 CFR Part 52 application and licensing process. These reviews can include the following:

   - Design documentation, such as design-specific HFE guidance documents and specifications
   - Prototype designs
   - Completed designs
   - HFE related inspections, tests, analyses, and acceptance criteria (ITAAC) (to ensure that an as-built plant conforms to the standard design certification)
   - HFE related design acceptance criteria (DAC) (to ensure that the applicant properly executes the design process after certification)

   For advanced NPPs (under 10 CFR Part 52), some HFE program elements may be deferred to the COL applicant. However, all HFE review criteria are addressed before plant startup.

2. *Review of the HFE Aspects of Control Room Modifications* - The NRC staff conducts reviews to ensure that voluntary modifications of existing NPPs are acceptable. This chapter can be used to review changes or modifications to the control room and other significant HSIs. Modifications may be extensive, such as a large-scale modernization of control room HSIs, using computer-based technology as part of a digital I&C upgrade program. Such a program can result in substantial modifications to alarms, controls, and displays that are associated with structures, systems, and components (SSCs) important to safety; thus a new or common-cause failure can be created that is not bounded by previous analyses or evaluations. Such a modification may be considered potentially significant to plant safety, per 10 CFR 50.59(c)(2). Additional guidance related to 10 CFR 50.59 is provided in Regulatory Guide (RG) 1.187, "Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments," (NRC, 2000) and Nuclear Energy Institute (NEI) publication 96-07, "Guidelines for 10 CFR 50.59 Implementation," (NEI, 2000).

3. *Review of the HFE Aspects of Modifications Affecting Risk-Important Human Actions* - The NRC staff reviews voluntary modifications to ensure they are acceptable. This chapter can also be used to review changes or modifications to licenses for nuclear power plants that include changes to human actions, e.g., a license amendment request. While HSI modernization may be a large-scale modification, even smaller-scale modifications may be risk-important, especially when they affect operator actions that are credited in the SAR. An HFE review is conducted if such a modification affects the role of personnel or the tasks they perform and is potentially significant to plant safety. Modifications affect the role or tasks of personnel if they impose new or different demands on them to operate or maintain the plant, or otherwise ensure safety. An example of such a modification would be substituting manual actions for automatic actions for performing design functions described in the updated final safety analysis report (UFSAR)

   A modification may be considered potentially significant to plant safety, per the criteria in 10 CFR 50.59(c)(2). Additional guidance related to 10 CFR 50.59 is provided in RG 1.187 (NRC, 2000) and Nuclear Energy Institute (NEI) publication 96-07, "Guidelines for 10 CFR 50.59 Implementation," (NEI, 2000).

B. <u>Graded Approach to Review</u>

The review methodology presented in this document is discussed generically. In its complete form as applied to the review of the HFE aspects of a new plant, the review process provides a comprehensive, detailed evaluation (see Section II.A). However, the level of staff review of an applicant's HFE design should reflect the unique circumstances of the review. In addition, staff reviews should also reflect risk-informed regulation and considerations. The NRC, the nuclear industry, and the public have moved to a broader consideration of risk in many activities associated with NPPs. Therefore, risk importance is taken into account when deciding which particular items to review and the depth of review necessary. This aspect of grading the review is discussed in Section II.C below.

To reflect the need to grade the review, this chapter provides detailed examples of graded review criteria for several reviews:

- Control room modifications (see Section II.B)
- Modifications affecting human actions of high risk importance (see Section II.C.2)

- Modifications affecting human actions of moderate risk importance (see Section II.C.3)
- Modifications affecting human actions of lower risk importance (see Section II.C.4)

Within these graded review criteria, the guidance is selectively applied to address the demands of each specific review. The areas of review to be given attention for an applicant's submittal are based on:

- An evaluation of the information provided by the applicant
- The similarity of the associated HFE issues to those recently reviewed for other plants
- The determination of whether items of special or unique safety significance are involved

C. Review Interfaces

The reviews conducted in this section should be coordinated with those of other SRP chapters and sections. Important review interfaces are described below.

1. Chapter 7, "Instrumentation and Controls." The Electrical and Instrumentation and Controls Branch (EICB) has primary responsibility for the review activities associated with Chapter 7. Descriptions of HSI components and characteristics addressed by the Chapters 7 and 18 reviews should be consistent. As appropriate, the review results of one chapter should be considered in the review activities for the other chapter.

2. Section 13.1.1, "Management and Technical Support Organization." The IEHB has primary responsibility for reviewing the corporate-level management and technical organizations of the applicant and its major contractors under Section 13.1.1. This section addresses the need for clearly defined management and organizational responsibilities with regard to HFE considerations in plant design. Chapter 18, under Acceptance Criteria, includes a comprehensive summary of management's role in ensuring that HFE is adequately considered in new plant design and in the modification of an existing plant. Thus, the reviews of Section 13.1.1 and Chapter 18 should be conducted in a coordinated manner.

3. Section 13.1.2-13.1.3, "Operating Organization." The IEHB has primary responsibility for reviewing specific staffing requirements under Section 13.1.2-13.1.3. In addition, Chapter 18 specifies a systematic analysis of staffing requirements that includes a thorough understanding of task requirements and applicable regulatory requirements. This analysis addresses the requirements from Section 13.1.2-13.1.3 as an input. Reviewers should ensure that staffing requirements addressed under Section 13.1.2-13.1.3 are properly considered in the Chapter 18 analysis.

4. Section 13.2, "Training." The IEHB has primary responsibility for the review of Section 13.2, which provides specific criteria for reviewing training programs for reactor operators in Section 13.2.1 and nonlicensed plant staff in Section 13.2.2. Chapter 18 contains an area of review titled "Training Program Development," which provides criteria for reviewing the process by which training programs are developed. It addresses the relationship between training development and the overall HFE design process. Thus, these reviews should be conducted in a coordinated manner. Topics from the SRP Chapter 18 area of review that are related to the review of Section 13.2 are cross-referenced.

5. Section 13.5, "Plant Procedures." The IEHB has primary responsibility for the review of

Section 13.5, which provides specific criteria for the content of administrative procedures under Section 13.5.1 and operating and maintenance procedures under Section 13.5.2. Chapter 18 contains an area of review titled "Procedure Development," which provides criteria for the review of the procedure development process rather than the actual procedures. Thus, these reviews should be conducted in a coordinated manner. Topics from the Chapter 18 review that are related to the review of Section 13.5 are cross-referenced.

6. Chapter 15, "Accident Analysis." Many branches have responsibility for the review of Chapter 15, which addresses anticipated operational occurrences and postulated accidents. Information from analyses conducted to address the criteria of Chapter 15 should be incorporated as input to the HFE design process.

7. Chapter 19, "Use of Probabilistic Risk Assessment in Plant-Specific, Risk-Informed Decisionmaking: General Guidance." The Probabilistic Safety Assessment Branch (SPSB) has primary responsibility for the review of SRP Chapter 19, which addresses probabilistic risk assessments for site-specific safety risks. The Chapter 18 review area "Human Reliability Analysis" addresses the relationship between HFE activities and probabilistic risk analysis/human reliability analysis (PRA/HRA) activities. Thus, these reviews should be conducted in a coordinated manner. Topics from the SRP Chapter 18 area of review that are related to the review of Chapter 19 are cross-referenced.

II. ACCEPTANCE CRITERIA

Acceptance is based upon conformance to the review criteria associated with the following areas of review.

A. Review of the HFE Aspects of a New Plant

A.1 HFE Program Management

The objective of this review is to confirm that the applicant has adequately considered the role of HFE and the means by which HFE activities are accomplished. The review should verify that:

- The applicant has identified plans to oversee design and construction of the nuclear facility in accordance with the requirements of 10 CFR 50.34(f)(3)(vii), as described in SRP Section 13.1.1, "Management and Technical Support Organization."
- The applicant has an HFE design team with the responsibility, authority, placement within the organization, and composition to ensure that the design commitment to HFE is achieved, as required by 10 CFR 50.34(f)(2)(iii).
- The team is guided by an HFE program plan to ensure the proper development, execution, oversight, and documentation of the HFE program.
- The overall HFE program appropriately considers and address the deterministic aspects of design, as discussed in RG 1.174

This HFE program plan should describe the technical program in sufficient detail to ensure that all aspects of the HSIs, procedures, and training are developed, designed, and evaluated on the basis of a structured top-down systems analysis using accepted HFE principles.

The applicant's HFE program management should be evaluated in accordance with the review criteria of NUREG-0711, "Human Factors Engineering Program Review Model."

A.2 Operating Experience Review

The objective of this review is to verify that the applicant has identified and analyzed HFE-related problems and issues in previous designs that are similar to the current design under review so that these problems and issues may be avoided in the development of the new design. This review should also ensure that positive features of previous designs are retained. The operating experience review (OER) should be evaluated in accordance with the review criteria of NUREG-0711 and should satisfy the requirements of 10 CFR 50.34(f)(3)(i).

A.3 Functional Requirements Analysis and Function Allocation

Functional requirements analysis is the identification and analysis of those functions that must be performed to satisfy plant safety objectives; that is, to prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public. Function allocation analysis is the analysis of requirements for plant control and the assignment of control functions to (1) personnel (e.g., manual control), (2) system elements (e.g., automatic control and passive, self-controlling phenomena), and (3) combinations of personnel and system elements (e.g., shared control, automatic systems with manual backup).

The objective of this review is to verify that (1) the plant's functions that must be performed to satisfy plant safety objectives have been defined, and (2) that the allocation of those functions to human and system resources has resulted in a role for personnel that takes advantage of human strengths and avoids human limitations. Functional requirements analysis and function analysis should be evaluated in accordance with the review criteria of NUREG-0711.

A.4 Task Analysis

Task analysis is the analysis of human performance demands that result from the allocation of functions to personnel and the identification of HSI characteristics needed to support personnel task accomplishment. The objective of this review is to ensure that the applicant's task analysis identifies the specific tasks that are needed for function accomplishment and their information, control, and task-support requirements. The task analysis should be evaluated in accordance with the review criteria of NUREG-0711.

A.5 Staffing and Qualifications

The objective of this review is to verify that the applicant has analyzed the requirements for the number and qualifications of personnel in a systematic manner that includes a thorough understanding of task requirements and applicable regulatory requirements. The applicant's staffing and qualifications analyses should be evaluated in accordance with the review criteria of NUREG-0711.

A.6 Human Reliability Analysis

Human reliability analysis (HRA) is an evaluation of the potential for and mechanisms of human error that may affect plant safety. The objectives of this review are to ensure that (1) the applicant has addressed human-error mechanisms in the design of the HFE aspects of the plant to minimize the likelihood of personnel error, and ensure errors are detected and recovered from; and (2) the HRA activity effectively integrates the HFE program and PRA. The applicant's HRA should be evaluated in accordance with the review criteria of NUREG-0711. In addition, the review should ensure that HRA activities performed in support of the HFE design are coordinated with PRA/HRA analyses required by 10 CFR 50.34(f)(1)(i) and addressed in Section 19.2 and other sections of the SRP.

A.7 Human-System Interface Design

The HSI design process represents the translation of function and task requirements into HSI characteristics and functions. The objective of this review is to evaluate the process by which HSI design requirements are developed and HSI designs are identified and refined. The review should ensure that the applicant has appropriately translated functional and task requirements to the detailed design of alarms, displays, controls, and other aspects of the HSI through the systematic application of HFE principles and criteria. The applicant's HSI design process should be evaluated in accordance with the review criteria of NUREG-0711, and the final design in accordance with the review criteria of NUREG-0700, "Human-System Interface Design

Review Guidelines."

A.8 Procedure Development

The objective of this review is to confirm that the applicant's procedure development program incorporates HFE principles and criteria, along with all other design requirements, to develop procedures that are technically accurate, comprehensive, explicit, easy to utilize, validated, and in conformance with 10 CFR 50.34(f)(2)(ii). Because procedures are considered an essential component of the HFE design, they should be a derivative of the same design process and analyses as the other components of the HSI (e.g., displays, controls, operator aids) and subject to the same evaluation processes. The applicant's procedure development program should be evaluated in accordance with the review criteria of NUREG-0711.

A.9 Training Program Development

The objective of this review is to ensure that the applicant has a systematic approach for the development of personnel training. The training development should include the following five activities:

- A systematic analysis of tasks and jobs to be performed
- Development of learning objectives derived from an analysis of desired performance following training
- Design and implementation of training based on the learning objectives
- Evaluation of trainee mastery of the objectives during training
- Evaluation and revision of the training based on the performance of trained personnel in the job setting

The training program should be developed in accordance with 10 CFR 50.120 and 10 CFR Part 55 to ensure that personnel's qualifications are commensurate with the performance requirements of their jobs. The applicant's training program should be evaluated in accordance with the review criteria of NUREG-0711 and should address applicable guidance provided in SRP Section 13.2, "Training."

A.10 Verification and Validation

Verification and validation (V&V) evaluations seek to comprehensively determine that the design conforms to HFE design principles and that it enables plant personnel to successfully perform their tasks to achieve plant safety and other operational goals. The applicant's V&V activities include operational condition sampling, design verification, integrated system validation, and human engineering discrepancy (HED) resolution. The objectives of the staff review of each of these activities are identified in the subsections below.

A.10.1 Operational Conditions Sampling

The applicant's sampling methodology identifies the range of operational conditions that guide V&V activities. The objectives of the review are to ensure that the applicant has identified a sample of operational conditions that (1) includes conditions that are representative of the range of events that could be encountered during operation of the plant, (2) reflects the characteristics that are expected to contribute to system performance variation, and (3) considers the safety significance of HSI components. The applicant's operational conditions sampling should be evaluated in accordance with the review criteria of NUREG-0711.

A.10.2 Design Verification

The applicant's verification ensures the design meets task and human requirements. Verification activities require a characterization of the HSI. The staff's review of design verification has the following objectives:

- Inventory and Characterization Review - The objective of this review is to ensure that the applicant's HSI inventory and characterization accurately describes all HSI displays, controls, and related equipment that are within the defined scope of the HSI design review.

- HSI Task Support Verification Review - The objective of this review is to ensure that the applicant verifies that the HSI provides all alarms, information, and control capabilities required for personnel tasks.

- HFE Design Verification Review - The objective of this review is to ensure that the applicant verifies that the characteristics of the HSI and the environment in which it is used conform to HFE guidelines.

The applicant's design verification should be evaluated in accordance with the review criteria of NUREG-0711.

A.10.3 Integrated System Validation

The objective of integrated system validation is to ensure that the integrated system design (i.e., hardware, software, and personnel elements) acceptably supports safe operation of the plant. Validation is based on performance-based tests. The applicant's design verification should be evaluated in accordance with the review criteria of NUREG-0711.

A.10.4 Human Engineering Discrepancy Resolution

HED resolution is the process of evaluating and resolving issues that are identified in V&V evaluations. The objectives of the staff's review are to ensure that the applicant's HED evaluation acceptably prioritizes HEDs in terms of their need for improvement and that design solutions and a realistic schedule for implementation are developed to address those HEDs selected for correction. The applicant's HED resolution should be evaluated in accordance with the review criteria of NUREG-0711.

A.11 Design Implementation

The objective of this review is to ensure that the applicant's as-built design conforms to the verified and validated design that resulted from the HFE design process. The applicant's design implementation should be evaluated in accordance with the review criteria of NUREG-0711.

A.12 Human Performance Monitoring

The objective of this review is to assure that the applicant has prepared a human performance monitoring strategy for ensuring that no significant safety degradation occurs because of any changes that are made in the plant and to provide adequate assurance that the conclusions that have been drawn from the evaluation remain valid over time. The applicant's performance monitoring strategy should be evaluated in accordance with the review criteria of NUREG-0711.

B. Review of the HFE Aspects of Control Room Modifications

License amendments involving major changes to the control room, such as control room modernization, should be reviewed using the guidance contained in Section II.A of this chapter. However, since the extent of such modifications can vary, the staff's review should be tailored using the additional guidance presented in this section.

B.1 HFE Program Management

The goals of the HFE program should address the need to consider the effects that the modification may have on the performance of personnel (as identified in NUREG-0711). The review should address the applications plan with respect to the following:

- Planning the installation to minimize disruptions to work
- Coordinating training and procedure modifications with implementing the modification to ensure that both accurately reflect the characteristics of the modification
- Conducting training to maximize personnel's knowledge of and skill with the new design before its implementation

B.2 Operating Experience Review

The operating experience of the plant being modified should be reviewed as part of the OER.

B.3 Functional Requirements Analysis and Function Allocation

Functional requirements analysis and function analysis should consider the following (as identified in NUREG-0711):

- Functional requirements analyses for modifications that are likely to change existing safety functions, introduce new functions for systems supporting safety functions, or involve unclear functional requirements that may be important to safety.

- Function allocation analyses for modifications that are likely to change the allocation between personnel and plant systems of functions important to safety.

- A change in an operator's role due to a modification should be examined within the context of its effects on the operator's overall responsibilities.

B.4 Task Analysis

The following considerations should be addressed in the review of plant modifications that are likely to affect human actions (HAs) previously identified as risk-important, cause existing HAs to become risk-important, or create new actions that are risk-important (as identified in NUREG-0711):

- The tasks analyses should be revised and updated to reflect requirements of the modification; the scope should include tasks involving the modification and its interactions with the rest of the plant, including those resulting from functions addressed in the analyses of functional requirements and function allocation. For maintenance, tests, inspections, and surveillances, attention should be given to risk-important actions that are new or supported by new technologies (e.g., new capabilities for online maintenance).

- The task analysis should identify the design characteristics of the existing HSIs that support the performance of experienced personnel (e.g., support high levels of performance during demanding situations).

B.5 Human-System Interface Design

The following considerations should be addressed in the review of design modifications (as identified in NUREG-0711):

- The extent to which HSI modifications are consistent with users' existing strategies
- The extent to which HSI modifications support crew coordination
- The degree to which the HSI reflects changes in the integration among plant systems

The final design modifications should be reviewed in accordance with the review criteria of NUREG-0700, as applicable.

B.6 Procedure Development

The review should evaluate whether procedures are modified and ensure their content, format, and integration accurately reflect changes in the plant, human actions, and HSIs (as identified in NUREG-0711).

B.7 Training Program Development

The review should evaluate whether any changes or increases in retraining are warranted following plant modernization programs (as identified in NUREG-0711).

B.8 Verification and Validation

1. Operational Conditions Sampling. Tasks that involve the modification should reflect the operational conditions (as discussed in NUREG-0711) and should address the potential effect of negative transfer of learning when the new and old components are different and impose different demands on personnel. The applicant's sampling should also consider any effects on performance of having both old and new versions of the same HSI components in place.

2. HSI Task Support Verification. HSI task support verification should focus on the HSIs that are relevant to the modification (as identified in NUREG-0711). For modifications to plant systems that do not include modifications of the HSIs, task support verification should identify any new demands for monitoring and control, and determine whether they are adequately addressed by the existing HSI design. HSIs for temporary

configurations and situations where both old and new HSIs are left in place should be evaluated for their potential to negatively impact performance.

3. HFE Design Verification. HFE design verification should focus on the HSIs that are relevant to the modification (as identified in NUREG-0711). HSIs for temporary configurations and situations where both old and new HSIs are left in place should be evaluated for their potential to negatively impact performance.

4. Integrated System Validation. The applicant should perform an integrated system validation for all modifications that may (as identified in NUREG-0711) (1) change personnel tasks; (2) change task demands, such as by changing task dynamics, complexity, or workload; or (3) interact with or affect HSIs and procedures in ways that may degrade performance. Integrated system validation may not be needed when a modification results in minor changes to personnel tasks such that they may reasonably be expected to have little or no overall effect on workload and the likelihood of error. The staff should ensure that the applicant validates that the functions and tasks allocated to plant personnel can be accomplished effectively when the integrated design is implemented. The applicant's test objectives and scenarios should be developed to address aspects of performance that are affected by the modification design, including personnel functions and tasks affected by the modification (as identified in NUREG-0711)

B.9 Design Implementation

The objective of this review is to ensure that the applicant's implementation of plant changes considers the effect on personnel performance and provides the necessary support to ensure safe operations. The applicant's design implementation should be evaluated in accordance with the review criteria of NUREG-0711. The following aspects of the design process should be addressed.

1. General Criteria. The staff's review should address whether the applicant can ensure that (as specified in NUREG-0711):

   - The reactor fuel is safely monitored during the shutdown time period while the physical modifications are being implemented in the control room.
   - Operations and maintenance crews are fully trained and qualified to operate and maintain the plant prior to starting up with the new systems and HSIs in place.
   - Modifications in plant procedures and training reflect changes in plant systems, crew roles and responsibilities, HSIs, and procedures for the new systems and HSIs should be in place prior to startup.
   - The applicant has a plan to monitor the initial phase of startup to identify and address any problems that arise.

2. Modernization Programs Consisting of Many Small Modifications. The staff's review should address whether the applicant can ensure that (as identified in NUREG-0711) each modification follows an HFE program that ensures standardization and consistency, and that modifications fulfill a clear operational need and do not interfere with existing systems.

3. Modernization Programs Consisting of Large Modifications During Multiple Outages. The staff's review should address whether the applicant can ensure that (as identified in

NUREG-0711):

- Task analysis is performed for each interim configuration to ensure that the task demands that are unique to interim configurations are known.
- HRA addresses any unique tasks that may affect risk or any changes to existing tasks due to the interim configuration.
- The HSIs needed to perform important tasks are consistent and standardized.
- Procedures are developed for temporary configurations of systems and HSIs that are used by personnel when the plant is not shut down.
- Training is developed for temporary configurations of systems, HSIs, and procedures that are used by personnel when the plant is not shut down.
- Temporary configurations are evaluated using V&V.

4. <u>Modernization Programs Where Both Old and New Equipment Are Left in Place</u>. The staff's review should address whether the applicant can ensure that (as identified in NUREG-0711) the potential for negative effects on personnel performance has been evaluated.

5. <u>Modernization Programs Where New Nonfunctional HSIs Are In Place In Parallel With Old Functional HSIs</u>. The staff's review should address whether the applicant can ensure that (as identified in NUREG-0711) the potential for negative effects on personnel performance due to control room or HSI clutter arising from having both old and new HSIs available in parallel is evaluated and that the nonfunctional state of the HSIs is clearly indicated.

C. <u>Review of the HFE Aspects of Modifications Affecting Risk-Important Human Actions</u>

The staff's review of license amendments and actions involving plant changes that affect important human actions (HAs) use a graded, risk-informed approach in conformance with Regulatory Guide (RG) 1.174 (NRC, 1998). The staff's review uses a two-phase approach. The first phase is a screening analysis to determine the risk associated with the plant modification and its associated HAs using both quantitative and qualitative information (see Section C.1 below). Plant modifications and HAs are categorized into regions of high, medium, and lower risk. This categorization is used to determine the level of HFE review needed. Changes that involve more risk-significant HAs receive a detailed review (see Section C.2 below), while those of moderate risk significance receive a less detailed review (see Section C.3 below). HAs in the lowest risk region receive minimal HFE review (see Section C.4 below).

C.1 Risk Screening

Applicants should evaluate the risk associated with the proposed modification and the HAs associated with it. The applicant's risk screening should be evaluated in accordance with the review criteria of "Guidance for the Review of Changes to Human Actions" (draft NUREG-1764, December 2002).

1. <u>Determine the Risk of the Entire Modification</u>. The first review step is to perform a risk-informed screening of the entire modification, including both equipment and HAs, in accordance with the review criteria of draft NUREG-1764, for both permanent and temporary changes. As part of this evaluation, the staff should determine whether the PRA information submitted as part of the risk-informed (R-I) submittal is suitable. The review criteria defined in RG 1.174 and SRP Chapter 19 should be used. If the staff determines that the information is not suitable, a generic method screening process

should be used (see item 4 below). RG 1.174 notes that licensee applications that lie in Region I are not normally permitted. If the entire modification is in Region I, the staff determines whether the modification is rejected. If it is rejected, then no additional HFE review is needed. If it is not rejected, the staff determines whether the modification contains only HAs or if it includes both equipment and HAs. If the modification contains only HAs (no equipment modifications) and was determined to be in Region I, then the HA should be reviewed using the Region I criteria in Section C.2 below. If the modification contains equipment and HAs, then the risk importance of the HA should be evaluated (see item 2 below).

2. Determine the Risk of the HAs. The second review step is to perform a risk-informed screening of the HA portion of the modification in accordance with the review criteria of draft NUREG-1764. This is done by evaluating both the risk achievement worth (RAW) and the Fussell-Vesely (FV) risk importance measures. HAs will be preliminarily sorted into the three Regions.

3. Perform Qualitative Screen of the HAs. The third risk-screening step is to identify whether there are qualitative factors that should be taken into account when determining the risk importance of the HA. This step may be used to adjust the review region either up or down. This evaluation should in accordance with the review criteria of draft NUREG-1764.

4. Review of Non-Risk-Informed Submittals. In keeping with RG 1.174, a licensee submittal to the NRC may be risk-informed (R-I) or not at the licensee's option. If it is not R-I, then the staff may choose to use the Generic Method to determine risk in accordance with the review criteria of draft NUREG-1764. The Generic Method will result in a proposed Region (I, II, or III) for the review. Qualitative screening is then applied to the proposed region to see if it needs to be adjusted. Alternatively, the staff may choose to perform a deterministic review without using the risk screening methodology. .

5. Determine the Level of HFE Review. Based on the quantitative and qualitative information available, the staff should classify the HA into one of three risk regions in accordance with the review criteria of draft NUREG-1764. Region I HAs, high risk, are reviewed using the criteria in Section C.2 below. Region II HAs, moderate risk, are reviewed using the criteria in Section C.3 below. Region III HAs, minimal risk, are reviewed using the criteria in Section C.4 below.

C.2 Region I HFE Review

HAs in the high-risk category should be reviewed using the Region I review criteria provided below.

1. General Deterministic Review Criteria. The applicant should provide adequate assurance that deterministic aspects of design, as discussed in RG 1.174, have been appropriately addressed. The staff should evaluated the deterministic aspects of the design, including that the change meets current regulations and does not compromise defense-in-depth, in accordance with the review criteria of draft NUREG-1764.

2. Operating Experience Review. The applicant should identify and analyze HFE-related problems and issues encountered previously in designs and human tasks that are

similar to the planned modification so that issues that could potentially hinder human performance can be addressed. The OER should address the operating histories of plant systems, HAs, procedures, and HSI technologies related to the proposed changes to HAs. The staff's evaluation should be conducted in accordance with the review criteria of draft NUREG-1764.

3. Functional Requirements Analysis And Functional Allocation. The applicant should define any changes in the plant's safety functions (functional requirements analysis), and provide evidence that the allocation of functions between humans and automatic systems provides an acceptable role for plant personnel; i.e., the allocations take advantage of human strengths and avoid functions that would be negatively affected by human limitations (functional allocation). The staff's review should addresses all plant functions affected by the change in HAs, including changes to the functions and to their allocation between personnel and automatic systems in accordance with the review criteria of draft NUREG-1764.

4. Task Analysis. The applicant should identify the behavioral requirements of the tasks personnel are required to perform. The task analysis should form the basis for specifying the requirements for the HSI, procedures, and training. The task analyses should address HAs in their entirety, including all pertinent plant conditions, situational factors, and performance-shaping factors. While the primary focus is licensed operator tasks, tasks performed by other personnel (e.g., maintenance, testing, inspection, and surveillance) that occur at the same time as the HAs and directly influence the actions are included in the task analysis. The staff should review the applicant's task analysis in accordance with the review criteria of draft NUREG-1764.

5. Staffing and Qualifications. The applicant should analyze the proposed change in HAs to determine the number and qualifications of personnel based on task requirements and applicable regulatory requirements. The analysis should addresses personnel requirements for all conditions in which the HA may be performed. The staffing and qualification review should be conducted in accordance with the review criteria of draft NUREG-1764.

6. Probabilistic Risk and Human Reliability Analysis. The applicant should (1) update the PRA model to reflect system, component, and HA changes that are necessary based on the proposed modification or HAs; (2) perform an analysis of the potential effects of the proposed changes upon plant safety and reliability, in a manner consistent with current, accepted PRA/HRA principles and practices, and (3) use the risk insights derived from the results in the selection of HAs and the development of procedures, HSI component lists, and training in order to limit risk and the likelihood of personnel error and to provide for error detection and recovery capability. The staff's HRA review should be conducted in accordance with the review criteria of draft NUREG-1764.

7. Human-System Interface Design. The applicant should translate function and task requirements into the detailed HSI design through the systematic application of HFE principles and criteria. The applicant's HSI design should be evaluated in accordance with the review criteria of draft NUREG-1764. The review should address the design of temporary and permanent modifications to the HSI, including new HSI components and the modification of existing ones, for the proposed changes in the HAs. Where changes in HAs result in modifications to large portions of the HSI or in the use of HSI technologies that do not have proven operating histories, the review may also examine

the HSI design process using the review criteria of NUREG-0711, Rev. 1. The review addresses aspects of the HSI and the work environment that affect the ability of the personnel to perform the HAs. The final design should be reviewed in accordance with the review criteria of NUREG-0700, as applicable.

8. Procedure Design. The applicant should modify applicable plant procedures and, where needed, provide guidance for the successful completion of the HAs. The procedures should adequately reflect changes in plant equipment and HAs. In the procedure development process, the applicant should apply HFE principles and criteria along with all other design requirements to develop procedure modifications that are technically accurate, comprehensive, explicit, easy to use, and validated. The applicant's procedure design should be evaluated in accordance with the review criteria of draft NUREG-1764.

9. Training Program Design. The applicant should develop and conduct adequate training for the HAs, including any changes in qualifications, as described in NRC Information Notice 97-78, "Crediting of Operation Actions In Place of Automatic Actions and Modification of Operator Actions, Including Response Times," (NRC, 1997). The training program should include all licensed and nonlicensed personnel who perform the changed HAs. The applicant's training program should be evaluated in accordance with the review criteria of draft NUREG-1764.

10. Human Factors Verification and Validation. The applicant should conduct V&V evaluations to (1) provide assurance that the HFE/HSI design provides all necessary alarms, displays, and controls to support plant personnel tasks (HSI task support verification); (2) provide assurance that the HFE/HSI design conforms to HFE principles, guidelines, and standards (HFE design verification); (3) provide adequate assurance that the HFE/HSI design can be effectively operated by personnel within all performance requirements applicable to the HA (integrated system validation); and (4) provide adequate assurance that the final product as built conforms to the verified and validated design that resulted from the HFE design process (final plant HFE/HSI design verification). The applicant's V&V should be evaluated in accordance with the review criteria of draft NUREG-1764.

11. Human Performance Monitoring Strategy. The applicant should have a human performance monitoring strategy to ensure that no adverse safety degradation occurs because of the changes that are made, to provide assurance that the conclusions that have been drawn from the evaluation remain valid over time, and to ensure that personnel have maintained the skills necessary to accomplish the assumed actions. The applicant's human performance monitoring strategy should be evaluated in accordance with the review criteria of draft NUREG-1764.

C.3 Region II HFE Review

HAs in the medium-risk category should be reviewed using the Region II review criteria provided below.

1. General Deterministic Review Criteria. The applicant should provide adequate assurance that deterministic aspects of design, as discussed in RG 1.174, have been appropriately addressed. The staff should evaluate the deterministic aspects of the design, including that the change meets current regulations and does not compromise

defense-in-depth, in accordance with the review criteria of draft NUREG-1764.

2. Analysis. The applicant should analyze the changes to the HA in terms of OER, functional and task analysis, and staffing and qualifications, and should identify HFE inputs for any modifications to the HSI, procedures, and training that may be necessary. The applicant's HFE analyses should be evaluated in accordance with the review criteria of draft NUREG-1764.

3. Design of HSIs, Procedures, and Training. The applicant should support the HA by appropriate modifications to the HSI, procedures, and training. The applicant's HSIs, procedures, and training design should be evaluated in accordance with the review criteria of draft NUREG-1764. Design modifications to the HSI should be reviewed in accordance with the review criteria of NUREG-0700.

4. Human Action Verification. The applicant should verify that the HA can be successfully accomplished with the modified HSI, procedures, and training. The applicant's verification should be evaluated in accordance with the review criteria of draft NUREG-1764.

C.4 Region III HFE Review

For an HA classified in third region, the staff review should verify that the action is, in fact, in Region III. Such a verification is accomplished by reviewing the licensee's analysis methods and risk results that show the placement of the action in that risk region. Typically no detailed HFE review is necessary. However, the staff may specify specific areas for review based on the results of the risk-screening process.

D. Technical Rationale

The NRC bases its HFE review on current regulatory requirements established in 10 CFR 50.34(g), "Conformance with the Standard Review Plan (SRP)," post-TMI bulletins and orders, and 10 CFR 50.34(f), "Additional TMI-Related Requirements." The NRC reviews HFE aspects of new control rooms (post-1982) to verify that they reflect "state-of-the-art human factors principles" as required by 10 CFR 50.34(f)(2)(iii) and that personnel performance is appropriately supported. For plants licensed under 10 CFR Part 52, the requirements of 10 CFR 50.34 are incorporated under 10 CFR 52.47. Meeting these requirements ensures that plant design, staffing, and operating practices provide assurance that plant safety will not be compromised by human error or deficiencies in human interfaces with hardware and software.

To support the review of an applicant's submittal for conformance to these 10 CFR requirements, the staff uses three primary guidance documents: NUREG-0700, NUREG-0711, and draft NUREG-1764. The technical basis upon which the staff's HFE review guidance was developed was (1) general systems theory and engineering principles; (2) available NPP industry HFE guidance, standards, guidance, and recommended practices developed in the industry (e.g., IEC and IEEE); and HFE guidance developed for complex systems in general (e.g., by groups such as DoD, NASA, and the Human Factors and Ergonomics Society). As part of the development process, the guidance and its associated technical reports were extensively reviewed by independent subject matter experts, professional organizations, and industry representatives. As a result the staff's guidance provides a technically valid basis upon which to review applicant HFE programs, processes, and designs.

NRC guidance for a systematic, top-down evaluation of HFE was originally provided in NUREG-0700, Revision 0. This document provided a methodology for the review of existing control rooms. It recommended that additional analyses be conducted for new control rooms to optimize the allocation of functions to humans and machines and further examine advanced control system technologies. Appendix B of NUREG-0700, Revision 0, was provided as one source of guidance regarding these analyses. The guidance of NUREG-0700, Revision 0, has been updated twice to reflect changes in HSI technologies.

NUREG-0711 addresses the integration of HFE in the design process and was originally developed to support NRC reviews of submittals for certification of new plant designs under 10 CFR Part 52. However, because it updates the guidance of Appendix B of NUREG-0700, Revision 0, it should be used for HFE reviews of new plant designs licensed under both 10 CFR Part 50 and 10 CFR Part 52. Portions of NUREG-0711 should also be used, as appropriate, to support the NRC in its reviews of upgrades of current control rooms.

Draft NUREG-1764, addresses the human performance aspects of changes to HAs that are credited for safety, especially those involving changes in the licensing basis of the plant; e.g., use of manual action in place of an automatic action for safety system operations. Risk-informed guidance and acceptance criteria are provided for the review of licensee proposals addressing such modifications. The review method uses a graded, risk-informed approach and provides guidance for reviewing the human performance aspects of changes to plant systems and operations. Three risk regions are defined: high, medium, and lower risk regions (called Regions I, II, and III). HAs are reviewed using human factors engineering criteria to ensure that the proposed HA can be reliably performed when called upon in the plant. HAs in the high-risk region receive a detailed review and those in the medium-risk region receive a less detailed review that is commensurate with their risk. For HAs falling into the lower-risk region, minimal (or no) human factors review is performed.

Thus, the HFE review process presented in this SRP chapter incorporates guidance from all three documents.

III. REVIEW PROCEDURES

The applicant should submit review materials for each review area. The general types of reports that the applicant may submit are described in NUREG-0711. These include:

1. Implementation Plan. This submittal describes the applicant's proposed methodology for meeting the acceptance criteria of a particular review element. An implementation plan review gives the applicant the opportunity to obtain staff review of and concurrence

in the applicant's approach before conducting the activities associated with the area. Such a review is desirable from the staff's perspective because it provides the opportunity to resolve methodological issues and provide input early in the analysis or design process when staff concerns can more easily be addressed than when the effort is completed.

2. Results Summary Report. This submittal describes the results of the applicant's efforts related to a particular review area. The NRC staff use the report as the main source of information for assessing the applicant's efforts using the review criteria contained in this document.

It is not intended that submittals necessarily be provided as separate reports. Rather it is important that information on methodology and results be available to the reviewer. In some cases an applicant may choose to provide this information in a single report or, in the case of license amendments, in the form of a safety analysis. It is also possible that, for more complex areas of review, such as HSI design or V&V, more than two reports may be submitted in order to address all review criteria. In addition to these reports, the reviewer may review sample work products (e.g., analyses and implemented designs).

In addition to the general reports, additional submittals are identified, where appropriate, in each HFE review area in NUREG-0711. The following are descriptions of special submittals and review considerations for specific areas of review:

1. HFE Program Management. The applicant should provide the following for staff review: HFE program plan describing the applicant's HFE goals/objectives, technical program to accomplish the objectives, a system to track HFE issues, the HFE design team, and the management and organizational structure to allow the technical program to be accomplished.

2. Operating Experience Review. The reviewer may also audit the issue tracking system for examination of OER issue treatment.

3. Human Reliability Analysis. The reviewers should review the PRA/HRA report(s) to gain a better understanding of the analysis method and results.

4. Human-System Interface Design. Other design-related HSI documents may be reviewed, such as applicant-developed guidance documents, detailed trade studies, technology assessments, or test/experiment reports developed to support the HSI design. In addition, a variety of mockups, prototypes, or similar physical representations of the HSI design may be available for preliminary review of the design implementation.

5. Procedure Development. Generic technical guidelines and sample procedures should be available for review.

6. Verification and Validation. The HFE issues tracking system, described in NUREG-0711, should be reviewed. The actual HSI design or a high-fidelity prototype or simulator of the HSI should be available for the staff to examine in conjunction with the verification reviews. In addition, the staff may witness the integrated system validation evaluations. A documented description of the final HSI design that resulted from the HSI task support verification, HFE design verification, integrated system validation, and issue resolution verification activities should be reviewed. Finally, the installation of the

completed design in the plant should be reviewed, if time and resources permit.

7. Human Performance Monitoring. Submittals for the staff's review of an applicant's human performance monitoring program should be made on a case-by-case basis.

8. ITAAC. For standard design certification reviews under 10 CFR Part 52, the procedures above should be followed, as modified by the procedures in SRP Section 14.3 and its subsections. SRP Section 14.3 contains procedures for the review of certified design material (CDM) for the standard design, including the site parameters, interface criteria, and inspections, tests, analyses, and acceptance criteria (ITAAC).

IV. EVALUATION FINDINGS

Acceptability of an individual area of review may be based on:

1. Satisfying all associated review criteria.

2. Demonstrating by alternative means that all review criteria have been satisfied. Alternative analysis methods proposed by the applicant must be acceptable to the NRC. In addition, the required amount of evidence may be reduced for some areas of review if it can be shown that the new design does not significantly differ from an accepted predecessor design and that no unresolved human factors issues exist.

3. Providing an acceptable justification for deviations from review criteria. Depending upon the review area and the nature of the deviation from review criteria, these justifications may be based upon such evidence as analyses of recent literature, analyses of current practices and operational experience, tradeoff studies, and the results of engineering experiments and evaluations.

An overall review conclusion is determined by comparing the goals of the HFE review, which are based on the type and purpose of the HFE review, to the evidence provided in the applicant's submittals. Important considerations include:

1. Were all relevant areas of review examined?

2. Was each area of review reviewed at the appropriate level (e.g., program description level, implementation plan level, and completed-area-of-review level)?

3. Were the findings for each area of review acceptable?

If the evidence provided by the review does not satisfy the goal of the HFE review, then additional analysis and design activities may be required of the applicant. These may include (1) additional analysis and review for areas the have not been examined at the completed-area-of-review level, (2) completion of the design or correction of design deficiencies identified through the review, and (3) appropriate testing or V&V.

V. IMPLEMENTATION

This SRP section will be used by the staff when performing safety evaluations of license applications submitted by applicants pursuant to 10 Part CFR 50 and 10 Part CFR 52. Except when the applicant proposes an acceptable alternative method for complying with specific portions of the Commission's regulations, the method described herein will be used by the staff in its evaluation of conformance with Commission regulations. It will also be used for evaluations of licensee-submitted requests for approval of HSI modifications (e.g., as contained in license amendment requests).

The provisions of this SRP section apply to review of applications docketed 6 months or more after the date of issuance of this section.

VI. REFERENCES

1. 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities."

2. 10 CFR Part 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants."

3. 10 CFR Part 55, "Operator's Licenses."

4. NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants."

5. NUREG-0696, "Functional Criteria for Emergency Response Facilities."

6. NUREG-0700, Revision 2, "Human-System Interface Design Review Guidelines."

7. NUREG-0711, Revision 1, "Human Factors Engineering Program Review Model."

8. NUREG-0737 and Supplement 1, "Clarification of TMI Action Plan Requirements."

9. NUREG-1342, "A Status Report Regarding Industry Implementation of Safety Parameter Display Systems."

10. NUREG-1764, (Draft), "Guidance for the Review of Changes to Human Actions."

11. NUREG/CR-2300, "PRA Procedures Guide: A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants."

12. NUREG/CR-2815, "Probabilistic Safety Analysis Procedures Guide."

13. NUREG/CR-3485, "PRA Review Manual."

14. NUREG/CR-6400, "HFE Insights for Advanced Reactors Based Upon Operating Experience."

15. Regulatory Guide 1.9, "Selection, Design, Qualification, and Testing of Emergency Diesel Generation Units Used as Class 1E Onsite Electric Power Systems at Nuclear Power Plants."

16. Regulatory Guide 1.22, "Periodic Testing of Protection System Actuation Functions."

17. Regulatory Guide 1.47, "Bypassed and Inoperable Status Indication for NPP Safety

Systems."

18. Regulatory Guide 1.62, "Manual Initiation of Protective Actions."

19. Regulatory Guide 1.81, "Shared Emergency and Shutdown Electrical Systems for Multi-Unit NPPs."

20. Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants To Assess Plant and Environmental Conditions During and Following an Accident."

21. Regulatory Guide 1.105, "Instrumentation Setpoints."

22. Information Notice 95-48, "Results of Shift Staffing Study."

23. Information Notice 97-78, "Crediting of Operator Action In Place of Automatic Actions and Modification of Operator Actions, Including Response Times."

ATTACHMENT 8

SRP Section 14.2.1
"Generic Guidelines for Extended Power Uprate Testing Programs"

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

14.2.1 GENERIC GUIDELINES FOR EXTENDED POWER UPRATE TESTING PROGRAMS

This Standard Review Plan (SRP) section provides general guidelines for reviewing proposed extended power uprate (EPU) testing programs. This review ensures that the proposed testing program adequately verifies that the plant can be operated safely at the proposed uprated power level.

Power uprates can be classified into three categories. Measurement uncertainty recapture power uprates are less than 2 percent and are achieved by implementing enhanced techniques for calculating reactor power. Stretch power uprates are typically up to 7 percent and do not generally involve major plant modifications. EPUs are greater than stretch power uprates and have been approved for increases as high as 20 percent. EPUs usually require significant modifications to major balance-of-plant equipment. A power uprate is classified as an EPU based on a combination of the proposed power increase and the plant modifications necessary to support the requested uprate. This SRP applies only to EPU license amendment requests.

REVIEW RESPONSIBILITIES

Primary -	Emergency Preparedness and Plant Support Branch (IEPB)
Secondary -	Reactor Systems Branch (SRXB)
	Plant Systems Branch (SPLB)
	Probabilistic Safety Assessment Branch (SPSB)
	Materials and Chemical Engineering Branch (EMCB)
	Electrical and Instrumentation & Controls Branch (EEIB)
	Mechanical & Civil Engineering Branch (EMEB)
	Reactor Operations Branch (IROB)

I. AREAS OF REVIEW

The Emergency Preparedness and Plant Support Branch coordinates the review of the overall power uprate testing program. Secondary review branches are responsible for reviewing EPU applications to ensure that the licensee has proposed an EPU testing program that demonstrates that structures, systems, and components (SSCs) will perform satisfactorily in service at the requested increased plant power level. Secondary review branches will assist IEPB in the review of proposed testing plans and acceptance criteria, as needed. The review of EPU testing programs should be performed in conjunction with staff reviews of other aspects of the EPU license amendment request.

II. ACCEPTANCE CRITERIA

Extended power uprate test program acceptance criteria are based on meeting the relevant requirements of the following regulations:

- Criterion 1, "Quality Standards and Records," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, as it relates to establishing the necessary testing requirements for SSCs important to safety, such that there is reasonable assurance that the facility can be operated without undue risk to the health and safety of the public.

- Criterion XI, "Test Control," of Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50, as it relates to establishment of a test program to assure that testing required to demonstrate that SSCs will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents.

- 10 CFR 50.90, "Application for Amendment of License or Construction Permit," as it relates to an application for an amendment following as far as applicable the form prescribed for original applications. Section 50.34, "Contents of Applications: Technical Information," specifies requirements for the content of the original operating license application, including that the Final Safety Analysis Report (FSAR) include plans for preoperational testing and initial operations.

Technical Rationale

This review ensures that the proposed EPU test program adequately demonstrates that SSCs will perform satisfactorily at EPU conditions. In particular, the EPU test program provides assurance that: (1) any power-uprate related modifications to the facility have been adequately constructed and implemented; and (2) the facility can be operated at the proposed EPU conditions in accordance with design requirements and in a manner that will not endanger the health and safety of the public.

The following paragraphs describe the technical rationale for application of the above acceptance criteria to the review of EPU test programs:

- Criterion 1 of Appendix A to 10 CFR Part 50, establishes the necessary testing requirements for SSCs important to safety; that is, SSCs that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public. Also, SSCs important to safety shall be designed, fabricated, erected and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability. Additionally, a quality assurance program shall be established to ensure that SSCs will satisfactorily perform their safety functions.

  Application of Criterion 1 of 10 CFR 50, Appendix A, to the EPU test program ensures testing is performed, as necessary, to provide assurance that SSCs continue to meet their original design specifications and capabilities. The quality assurance program ensures proper documentation and traceability that applicable testing was accomplished, and codes and standards were satisfied.

Throughout this SRP section, the term "important to safety" is used to refer to those SSCs to which this EPU testing guidance applies. Generic Letter (GL) 84-01, "NRC Use of the Terms 'Important to Safety' and 'Safety Related'," indicates that the term "important to safety" generally refers to plant equipment needed to meet the provisions of the General Design Criteria. However, as discussed in Section 2.1.5.6 of LIC-100, "Control of Licensing Basis for Operating Reactors," the General Design Criteria (GDC) are not applicable to plants with construction permits issued before May 21, 1971. Each plant licensed before the GDC were formally adopted was evaluated on a plant-specific basis, determined to be safe, and licensed by the Commission. For the purposes of the EPU test program review, the SSCs considered to be important to safety should be based on the plant-specific licensing basis.

- Criterion XI of Appendix B to 10 CFR Part 50 requires that a test program be established to assure that all testing required to demonstrate that SSCs will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents. The test program requirements include, as appropriate, proof tests prior to installation, preoperational tests, and operational tests of SSCs. Test procedures are required to include provisions for assuring that all prerequisites for the given test have been met, that adequate test instrumentation is available and used, and that the test is performed under suitable environmental conditions. Test results are required to be documented and evaluated to assure that test requirements have been satisfied.

  Application of Criterion XI of 10 CFR Part 50, Appendix B, to the EPU test program ensures that SSC capabilities to perform specified functions are not adversely impacted by increasing the maximum allowed power level. This also ensures that deficiencies are identified and corrected, and that testing activities are conducted in a manner which minimizes operational reliance on untested safety functions. This provides a high degree of assurance of overall plant readiness for safe operation within the bounds of the design and safety analyses, assurance against unexpected or unanalyzed plant behavior, and assurance against safety functional failures in service. Regulatory Guide (RG) 1.68, "Initial Test Programs for Water-Cooled Nuclear Power Plants," Revision 2, describes the general scope and depth of initial test programs that the NRC staff found acceptable during the review of original operating license applications. The SSCs subject to initial testing performed safety functions that included fission product containment; reactivity monitoring and control; reactor safe shutdown (including maintaining safe shutdown); core cooling; accident prevention; and consequence mitigation as specified in the design and credited in safety analyses.

- 10 CFR 50.90, "Application for Amendment of License or Construction Permit," requires that each licensee submitting a license amendment request fully describe the changes desired and follow, as far as practicable, the form prescribed for the original application. Section 50.34, "Contents of Applications: Technical Information," specifies requirements for the original operating license application. In particular, 10 CFR 50.34(b)(6)(iii) requires that each application for a license to operate a facility include in the FSAR plans for preoperational testing and initial operations. The initial test program (which includes preoperational testing and testing during initial operation) verifies that SSCs are capable of performing their safety functions as specified in the design and credited in safety analyses.

Application of 10 CFR 50.90 and 10 CFR 50.34(b)(6)(iii) to the EPU test program ensures that the licensee submits adequate information, commitments, and plans demonstrating that the requested power uprate does not invalidate testing requirements contained in the original licensing basis. Preoperational and initial startup testing invalidated by the requested increase in power level are evaluated and reperformed as necessary to demonstrate safe operation of the plant. This ensures that operation at the requested higher power level will be within the bounds of the design and safety analyses and that EPU testing activities will be conducted in a sequence and manner which minimizes operational reliance on untested SSCs or safety functions.

III. REVIEW PROCEDURES

The purpose of this review is to ensure that the proposed EPU testing program adequately controls the initial power ascension to the requested EPU power level. The EPU test program shall include sufficient steady-state and transient performance testing to demonstrate that SSCs will perform satisfactorily at the requested power level. The proposed EPU test program should be based on a systematic review of the initial plant test program to identify initial licensing power-ascension testing that may be invalidated by the requested EPU. Additionally, the EPU test program should include sufficient testing to demonstrate that EPU-related plant modifications have been adequately implemented.

A. Comparison of Proposed EPU Test Program to the Initial Plant Test Program

1. General Discussion

The licensee should provide a comparison of the proposed EPU testing program to the original power-ascension test program performed during initial plant licensing. The scope of this comparison shall include (1) all initial power-ascension tests performed at a power level of equal to or greater than 80 percent of the original licensed thermal power level; and (2) initial test program tests performed at lower power levels if the EPU would invalidate the test results. The licensee shall either reperform initial power-ascension tests within the scope of this comparison or adequately justify proposed deviations.

2. Specific Acceptance Criteria

Within its associated technical discipline, each secondary branch reviewer will determine if the licensee has adequately identified the following in the EPU license amendment request:

- all initial power-ascension tests performed at a power level of equal to or greater than 80 percent of the original licensed thermal power level;

- all initial test program tests performed at power levels lower than 80 percent of the original licensed thermal power level that would be invalidated by the EPU; and,

- differences between the proposed EPU power-ascension test program and the portions of the initial test program identified by the previous criteria.

The reviewer should refer to the plant-specific testing identified in FSAR Chapter 14.2, "Initial Plant Test Program" (or the equivalent FSAR section for non-standard format plants), and startup test reports, if available, to verify that the licensee has adequately identified the scope of the initial plant test program. Additionally, Attachment 1, "Typical Steady-State Power Ascension Testing Applicable to Extended Power Uprates," and Attachment 2, "Typical Transient Testing Applicable to Extended Power Uprates," to this SRP section provide a generic summary of power-ascension tests performed at or near full power.

If the licensee's proposed EPU test program does not include performance of testing originally performed during the initial plant test program, the reviewer shall ensure that the licensee adequately justifies all differences. The reviewer should refer to Section III.C, below, for guidance on assessing the adequacy of justifications for proposed differences.

B. Post Modification Testing Requirements for SSCs Important to Safety Impacted by EPU-Related Plant Modifications

1. General Discussion

EPUs usually require significant modifications to major balance-of-plant equipment, in addition to setpoint and operating parameter changes. Therefore, within its respective technical area, each secondary review branch will assess if the licensee adequately evaluated the aggregate impact of EPU plant modifications, setpoint adjustments, and parameter changes that could adversely impact the dynamic response of the plant to anticipated initiating events. The objective of this review is to verify that the licensee has proposed a testing program which demonstrates that EPU-related modifications to the facility have been adequately implemented.

The reviewer is not expected to evaluate the specific component- and system-level testing requirements for each plant modification, parameter change, or setpoint adjustment. Based on previous experience, testing required by Technical Specifications and existing 10 CFR 50, Appendix B, quality assurance programs have been adequate to demonstrate individual system or component performance characteristics. Additionally, the scope of power ascension testing described in RG 1.68, Appendix A, Section 5, "Power Ascension Tests," is generally limited to tests which demonstrate that the facility operates in accordance with design both during normal steady-state conditions, and, to the extent practical, during and following anticipated operational occurrences. Anticipated operational occurrences are those conditions of normal operation which are expected to occur one or more times during the life of the plant and include events such as loss of all offsite power, tripping of the main turbine generator set, and loss of power to all recirculation pumps. Therefore, this review is intended to ensure that plant equipment important to safety that supports functions that rely on the integrated operation of multiple SSCs following an anticipated operational occurrence are adequately demonstrated prior to extended operation at the requested EPU power level.

2. Specific Acceptance Criteria

Based on review of the licensee's EPU license amendment request, the reviewer will determine if the licensee has adequately identified the following:

- plant modifications and setpoint adjustments necessary to support operation at power uprate conditions, and

- changes in plant operating parameters (such as reactor coolant temperature, pressure, T_{ave}, reactor pressure, flow, etc.) resulting from operation at EPU conditions.

The reviewer should assess if the licensee adequately identified SSCs important to safety that are affected by EPU-related modifications, setpoint adjustments, and changes in plant operating parameters. In particular, the licensee should have considered the impact of first-of-a-kind plant modifications, the introduction of new system dependencies or interactions, and changes in system response on the capability of plant equipment to perform their specified functions. The review scope can be limited to those SSCs important to safety that are used to mitigate anticipated operational occurrences described in the plant-specific licensing basis. To assist in this review, Attachment 2 includes a listing of general transient testing acceptance criteria associated with typical anticipated operational occurrences.

The reviewer should verify that the proposed EPU test program adequately demonstrates the performance of SSCs important to safety that meet all of the following criteria: (1) the performance of the SSC is impacted by EPU-related modifications, (2) the SSC is used to mitigate an anticipated operational occurrence described in the plant-specific licensing basis, and (3) the SSC supports a function that relies on the integrated operation of multiple systems and components. If an SSC important to safety that meets these criteria cannot be adequately tested by overlapping individual component- or system-level tests, the licensee should propose suitable plant-level functional testing.

C. Justification for Elimination of EPU Power-Ascension Tests

1. General Discussion

The licensee may propose a test program that does not include all of the power-ascension testing that would normally be considered for inclusion in the EPU test program by the review criteria of Sections III.A and III.B, above. In these cases, the licensee shall provide an adequate justification for each of the power-ascension tests identified by the above review criteria that is not included in the EPU test program. For each proposed test exception within its technical area, each secondary review branch will verify the adequacy of the licensee's justification.

2. Specific Acceptance Criteria

If the licensee proposes not to perform a power-ascension test that would normally be considered for inclusion in the EPU test program by the review criteria contained in Sections III.A and III.B, above, the reviewer should ensure that the licensee provides an adequate justification. The proposed EPU test program shall be sufficient to adequately demonstrate that SSCs will perform satisfactorily in service. The reviewer should consider the following factors when assessing the adequacy of the licensee's justification:

a. Previous Operating Experience

If the licensee proposes not to perform a specified transient test based on operating experience, the reviewer should determine the applicability of the operating experience to the specific plant requesting the EPU. The reviewer should consider similarities in plant design and equipment; operating power level; test specifications and methods; and operating and emergency operating procedures.

b. Introduction of New Thermal-Hydraulic Phenomena or Identified System Interactions

The reviewer should ensure that the licensee adequately addressed the effects of any new thermal-hydraulic phenomena or system interactions that may be introduced as a result of the EPU.

c. Facility Conformance to Limitations Associated With Analytical Analysis Methods

If the basis for elimination of a power ascension test from the EPU test program relies on the use of analytical analysis methods, the licensee's justification should include consideration of the facility's conformance to limitations associated with analytical analysis methods. These limitations may include, but are not limited to, plant operating parameters, system design and configuration, and reactor power level.

d. Plant Staff Familiarization With Facility Operation and Trial Use of Operating and Emergency Operating Procedures

Plant modifications and parameter changes, in conjunction with increased decay heat generation associated with higher power operation, can impact the execution of abnormal and emergency operating procedures. For example, the EPU may change the timing and sequence of significant operator actions used in abnormal and emergency operating procedures, or could impact accident mitigation strategies in abnormal or emergency operating procedures.

For each EPU license amendment request, the technical branch responsible for operator licensing and human performance reviews the impact of the requested power uprate on operator training and human factors in accordance with separate standard review plan guidance. These reviews include an evaluation of the changes in operator actions, procedures, and training (including necessary changes to the control room simulator) resulting from the EPU. Although the initial power-ascension test program objectives, as described in Reference 8, includes plant staff familiarization with facility operation and trial use of plant abnormal and emergency operating procedures; based on previous review experience, it is not expected that power-ascension testing would normally be performed solely for the purposes of procedure verification or operator familiarization. However, if the review of the operator training and human factors aspects of the EPU indicates the need to perform power-ascension testing for the purposes of procedure verification or operator familiarization, the EPU test program review shall be coordinated with the technical branch responsible for operator licensing and human performance.

e. Margin Reduction in Safety Analysis Results for Anticipated Operational Occurrences

The licensee's justification for not performing a particular power-ascension test may include a consideration of the change in the associated safety analysis results due to the proposed EPU. To aid in this review, the information provided in Attachment 2 to this SRP section includes a reference to the safety analysis SRP sections related to typical power ascension transient tests, as applicable. For safety analysis acceptance criteria that can be quantitatively measured (e.g. peak reactor coolant system pressure), a reduction in available margin by less than approximately 10 percent would normally be considered to be a minimal change in consequences. The available margin is the difference between the standard review plan accident analysis acceptance criterion of interest and the plant-specific value calculated at EPU conditions. For larger reductions in available margin, the licensee may consider such factors as the amount of remaining margin; the sensitivity of the results to changes in analysis assumptions; and the capability of transient testing to provide useful confirmatory data.

Although the initial power-ascension test program objectives, as described in Reference 8, included validation of analytical models and verification of assumptions used for predicting plant response to anticipated transients and postulated accidents, transient testing is not required for the purposes of analytical code validation for EPU license amendment reviews. The applicability and use of accident analysis analytical codes for an EPU is reviewed by the staff in accordance with separate standard review plan guidance.

f. Guidance Contained in Vendor Topical Reports

The NRC previously reviewed and accepted General Electric (GE) Company Licensing Topical Report, "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate" (referred to as ELTR-1), NEDC-32424P-A, Class III, February 1999, as a review basis for EPU license amendment requests. This topical report provided specific guidance for the performance of integrated system transient testing at EPU conditions. As described in Section 5.11.9.d and Appendix L.2.4 of ELTR-1, the generator load rejection and the main steam isolation valve (MSIV) closure tests verify that the plant performance is as predicted and projected from previous test data.

On March 31, 2003, the NRC approved the use of GE licensing technical report NEDC-33004P, "Constant Pressure Power Uprate," for constant pressure power uprate (CPPU) EPU licensing applications. However, as noted in the staff's safety evaluation, the staff did not accept the elimination of large transient testing (e.g. the MSIV closure and turbine generator load rejection described in NEDC-32424P-A) from the scope of the CPPU test program. The staff noted that the need to conduct large transient testing for a CPPU would be considered on a plant-specific basis.

For PWRs, Westinghouse Report WCAP-10263, "A Review Plan for Uprating the Licensed Power of a Pressurized Water Reactor Plant," provides limited guidance for power uprate testing. Specifically, the document states that the recommended test program for the nuclear steam supply system and interfacing balance-of-plant systems be developed on a plant-specific basis depending on the magnitude of hardware modifications and the magnitude of the power uprate.

Although the NRC has previously approved certain exceptions to power-ascension testing requirements, the reviewer should assess the licensee's proposed justifications on a plant-specific basis.

g. Risk Implications

For cases where the licensee proposes a risk-informed basis for not performing certain transient tests, SPSB should be consulted to assist in the review. Risk-informed justifications for not performing transient tests should be carefully weighed against the potential benefits of performing the testing. In addition to the risks inherent in initiating a plant transient, the review should also consider the benefit of identifying potential latent equipment deficiencies or other plant problems under controlled circumstances during transient testing. In any case, a risk-informed justification should not be used as the sole basis for not performing transient testing.

If the licensee provides adequate justification for not performing certain power-ascension tests, the staff may conclude that the EPU test program is acceptable without the performance of these tests.

D. Evaluate the Adequacy of Proposed Testing Plans

1. General Discussion

The EPU amendment request should include plans for the initial approach to the increased EPU power level and steady-state testing that will be used to verify that the reactor plant operates within design parameters.

2. Specific Acceptance Criteria

For each EPU power-ascension test proposed by the licensee to demonstrate that the plant can be safely operated at EPU conditions, the staff will review the test objectives, summary of prerequisites and test methods, and specific acceptance criteria to establish that the functional adequacy of SSCs is verified. This review assures that the test objectives, test methods, and the acceptance criteria are acceptable and consistent with the licensing basis for the facility.

Each secondary review branch will review the licensee's plans for the EPU test program within its respective technical area. The licensee's EPU test program should include the following:

- The initial approach to the uprated EPU power level should be performed in an incremental manner and include steady-state power hold points to evaluate plant performance above the original full-power level.

- The licensee should propose appropriate testing and acceptance criteria to ensure that the plant responds within design predictions. The predicted responses should be developed using real or expected values of items such as beginning-of-life core reactivity coefficients, flow rates, pressures, temperatures, and response times of equipment and the actual status of the plant, and not the values or plant conditions used for conservative evaluations of postulated accidents.

- Contingency plans should be implemented if the predicted plant response is not obtained.

- The test program should be scheduled and sequenced to minimize the time untested SSCs important to safety are relied upon during operation above the original licensed full-power level. Safety-related SSCs relied upon during operation shall be verified to be operable in accordance with existing Technical Specification and Quality Assurance Program requirements.

To assist this review, Attachments 1 and 2 to this SRP section provide a generic listing of full power steady-state and transient tests and related acceptance criteria that are potentially applicable to an EPU test program.

If a power-ascension test is needed to demonstrate that the plant can be operated safely at EPU conditions, the reviewer shall determine if a license condition should be imposed to ensure that this testing is performed in a timely and controlled manner.

IV. EVALUATION FINDINGS

When the review of the information in the EPU amendment application is complete and the reviewer has determined that it is satisfactory and in accordance with the acceptance criteria in Section II above, a statement similar to the following should be provided in the staff's Safety Evaluation Report (SER):

"The staff has reviewed the EPU test program in accordance with SRP Section 14.2.1. This review included an evaluation of: (1) plans for the initial approach to the proposed maximum licensed thermal power level, including verification of adequate plant performance; (2) transient testing necessary to demonstrate that plant equipment will perform satisfactorily at the proposed increased maximum licensed thermal power level; and (3) the test program's conformance with applicable regulations. The staff concludes that the proposed EPU test program provides adequate assurance that the plant will operate in accordance with design criteria and that SSCs affected by the EPU or modified to support the proposed power increase will perform satisfactorily in service. Further, the staff finds that there is reasonable assurance that the EPU testing program satisfies the requirements of 10 CFR Part 50, Appendix B, Criterion XI, 'Test Control.' Therefore, the NRC staff finds the proposed EPU test program acceptable."

V. IMPLEMENTATION

This SRP section will be used by the staff when performing safety evaluations of EPU license amendment applications submitted pursuant to 10 CFR 50.90. This SRP is not intended to be used in place of plant-specific licensing bases to assess the acceptability of an EPU application. Applicability of this SRP is determined on a plant-specific basis consistent with the licensing basis of the plant.

In addition, where the NRC has approved a specific methodology (e.g., topical report) for the type of power uprate being requested, licensees should follow the format prescribed for that specific methodology and provide the information called for in that methodology and the NRC's letter and safety evaluation approving the methodology. Except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method described herein will be used by the staff in its evaluation of conformance with Commission regulations.

VI. REFERENCES

1. 10 CFR Part 52, §52.47 "Contents of Applications."

2. 10 CFR Part 50, Appendix B, Criterion XI, "Test Control."

3. NUREG-1503, "Final Safety Evaluation Report Related to the Certification of the Advanced Boiling Water Reactor," Volumes 1 and 2, July 1994.

4. SECY-01-0124, "Power Uprate Application Reviews," dated July 9, 2001. The related Staff Requirements Memorandum is dated May 24, 2001.

5. General Electric Company Licensing Topical Report, "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate" (ELTR-1), NEDC-32424P-A, Class III, February 1999.

6. General Electric Company Licensing Topical Report, "Generic Evaluations of General Electric Boiling Water Reactor Extended Power Uprate," (ELTR-2), NEDC-32523P-A, Class III, February 2000, and Supplement 1, Volumes I and II.

7. General Electric Company Licensing Topical Report, "Constant Pressure Power Uprate," NEDC-33004P, Revision 1, July 2001.

8. NRC Regulatory Guide 1.68, "Initial Test Programs for Water-Cooled Nuclear Power Plants," Revision 2, August 1978.

9. NRR Office Instruction LIC-100, "Control of Licensing Basis for Operating Reactors."

10. NRR Office Instruction LIC-101, "License Amendment Review Procedures."

11. NRR Office Instruction LIC-500, "Processing Requests for Reviews of Topical Reports."

12. Westinghouse WCAP-10263, "A Review Plan for Uprating the Licensed Power of a Pressurized Water Reactor Power Plant," January 1983.

13. NRC Inspection Manual, Part 9900, "10 CFR Part 50.59, Changes, Tests and Experiments," Change Notice Number 01-008.

14. NRC Information Notice 2002-26, "Failure of Steam Dryer Cover Plate After a Recent Power Uprate," September 11, 2002.

Typical Steady-State Power Ascension Testing Applicable to Extended Power Uprates

Power Ascension Test	Reference	Recommended Initial Conditions	Typical Test Acceptance Criteria	Primary Technical Review Branch
Conduct vibration testing and monitoring of reactor vessel internals and reactor coolant system components	Regulatory Guide (RG) 1.68, App A 4.s, 5.p	Lowest practical power level	Reactor vessel and reactor coolant system component vibration characteristics within design. (See NRC Information Notice 2002-26 and RG 1.20)	EMEB
Measure power reactivity coefficients (PWR) or power vs. flow characteristics (BWR)	RG 1.68, App A 5.a	100% of rated thermal power (RTP)	Characteristics in accordance with design	SRXB
Steady-state core performance	RG 1.68, App A 5.b	100% of RTP	Characteristics in accordance with design	SRXB
Control rod patterns exchange	RG 1.68, App A 5.c	Power equal to highest power level that rod exchanges will be allowed at power	Core limits not exceeded	SRXB
Control rod misalignment testing	RG 1.68, App A 5.i	100% of RTP Rod misalignment equal to or less than TS limits	Demonstrate ability to detect misalignment	SRXB
Failed fuel detection system	RG 1.68, App A 5.q	100% of RTP	Verify proper operation	IEPB
Plant process computer	RG 1.68, App A 5.r	100% of RTP	Inputs and calculation are correct	SPLB/EEIB
Calibrate major or principal plant control systems	RG 1.68, App A 5.s	100% of RTP	Verify performance	SRXB/SPLB
Main steam and main feedwater system operation	RG 1.68, App A 5.v	100% of RTP	Operate in accordance with design performance requirements	SPLB
Shield and penetration cooling systems	RG 1.68, App A 5.w	100% of RTP	Maintain temperature within design limits	SPLB
Engineered Safety Features (ESF) auxiliary and environmental systems	RG 1.68, App A 5.x	100% of RTP	Capable of performing design functions	SPLB
Calibrate systems used to determine reactor thermal power	RG 1.68, App A 5.y	100% of RTP	Verify performance	EEIB
Chemical and radiochemical control systems	RG 1.68, App A 5.a.a	100% of RTP	Control systems function in accordance with design	IEPB
Sample reactor coolant system and secondary coolant systems	RG 1.68, App A 5.a.a	100% of RTP	Chemistry limits are not exceeded	EMCB

Power Ascension Test	Reference	Recommended Initial Conditions	Typical Test Acceptance Criteria	Primary Technical Review Branch
Radiation surveys	RG 1.68, App A 5.b.b	100% of RTP	Shielding adequacy and identify 10 CFR Part 20 high-radiation zones	IEPB
Ventilation systems (including primary containment and steam line tunnel)	RG 1.68, App A 4.j and 5.f.f	100% of RTP	Maintain service areas within design limits	SPLB
Acceptability of reactor internals, piping, and component movement, vibrations, and expansions	RG 1.68, App A 1.a.1, 1.a.3, 1.e., and 5.o.o	Lowest practical power level	Parameters within design values	EMEB

Typical Transient Testing Applicable to Extended Power Uprates

Transient Test	Reference	Typical Reactor Plant Initial Conditions	Typical Transient Test Acceptance Criteria	Associated Standard Review Plan Accident Analyses Section (as applicable)
Dynamic response of plant to loss of feedwater flow	RG 1.68, Appendix A, Section 5 (Introduction)		Plant performance in accordance with design	15.2.7 Loss of Normal Feedwater Flow
Relief valve testing	RG 1.68, App A 4.p and 5.t Inspection Procedure (IP) 72510	Reactor power level at predetermined power level plateaus All relief valves set in auto Individual valve functional tests at prescribed power level plateaus Individual valve capacity tests at low power (25% of RTP) using bypass valve movement or turbine generator output as a measurement variable	Relief valve rating at a specified pressure setting Delay time between the signal initiating relief valve opening and the start of motion Opening stroke time of the main valve disc and distance Closing stroke time of the main valve piston following release of the pneumatically operated mechanical push rod	15.1.2 Inadvertent Opening of a Steam Generator Relief or Safety Valve 15.6.1 Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve or a BWR Pressure Relief Valve
Dynamic response of plant to design load swings	RG 1.68, App A 5.h.h	100% of RTP	Performance in accordance with design	
Dynamic response of plant to limiting reactor coolant pump trips or closure of reactor coolant system flow control valves (Reactor coolant recirculation pump trip test)	RG 1.68, App A 5.i.i IP 72512	100% of RTP Trip from steady-state power operation Recording of transients following trip and during pump restart Recording of limiting heat transfer parameters Return to two-pump operation in accord with facility operating procedures Trip of a single pump and of both pumps simultaneously.	Performance in accordance with design: Instrumentation is adjusted to provide an accurate conversion of individual jet pump Δp values to a summed core flow over the range of two-pump operations Recirculation pump instrumentation is calibrated Loop flow from single-tap and double-tap pumps agrees within 3% Core flow from single-tap and double-tap pumps agrees within 2% Individual jet pump flow variation from average pump flow is limited.	15.3.1 (BWR) & 15.3.2 (PWR) Loss of Forced Reactor Coolant Flow Including Trip of Pump Motor
Dynamic response of the plant to loss of feedwater heaters that results in most severe feedwater temperature reduction	RG 1.68, App A 5.k.k	90% of RTP	Performance in accordance with design	15.1.1 Decrease in Feedwater Temperature

Transient Test	Reference	Typical Reactor Plant Initial Conditions	Typical Transient Test Acceptance Criteria	Associated Standard Review Plan Accident Analyses Section (as applicable)
Dynamic response of plant to turbine trip (Turbine trip or generator trip)	RG 1.68, App A 5.l.l IP 72580 IP 72514	Trip from steady state operation at greater than 95% of RTP Initiation of the test by trip of the main generator output breaker Recirculation system flow control mode must be specified	Performance in accordance with design, including: Reactor coolant pumps do not trip Pressurizer spray valve opens and closes at the specified values Reactor pressure remains below the setpoint of the first safety valves, pressurizer safety valves do not lift or weep Pressurizer level within prescribed limits Steam system power actuated pressure relief valve opens and closes at specified values Reactor coolant pressure/temperature relationship remains within defined values Steam generator level remains within prescribed limits, no flooding of the steam lines during the transient, no initiation of ECCS and MSIV isolation during the transient Turbine bypass system operates to maintain specific pressure (plants with 100% bypass capability shall remain at power without scram during the transient) Plants with select-rod-insertion shall maintain power without scram from recirculation pump overspeed or cold feedwater effect Reactor protection system functions should be verified All safety and emergency core cooling systems such as the reactor protection system (RPS), high pressure coolant injection (HPCI), diesel generators, and reactor core isolation cooling (RCIC) function without manual assistance if called upon Normal reactor cooling systems should maintain adequate cooling and prevent actuation of automatic depressurization system, even though relief valves may function to control pressure Plant electrical loads (transferred as designed) Turbine overspeed criteria met	15.2.1 Turbine Trip
Dynamic response of plant to automatic closure of all main steam isolation valves	RG 1.68, App A 5.m.m IP 72510	Initial power level of 100% of RTP	Performance in accordance with design Acceptance criteria include MSIV closing time	15.2.4 Main Steam Isolation Valve Closure (BWR)

Transient Test	Reference	Typical Reactor Plant Initial Conditions	Typical Transient Test Acceptance Criteria	Associated Standard Review Plan Accident Analyses Section (as applicable)
Dynamic response of plant for full load rejection (Loss of Offsite Power Testing)	RG 1.68, App A 5.n.n IP 72517 IP 72582	100% of RTP with electrical system aligned for normal full-power operation and load rejection method should subject turbine to maximum credible overspeed condition Steady-state plant operations with greater than 10% generator output (IP 72517 & 72582). Trip of the plant with breakers in specified positions so that plant loads will be transferred directly to the diesel generators following loss of house power Recirculation system flow control mode specified	Performance in accordance with design, including: Automatic transfer of plant loads as designed, automatic start of diesel generators, automatic load of diesel generators in the specified sequence Reactor pressure remains below the first safety valve setting. Pressurizer safety valves do not lift All safety systems such as RPS, HPCI, diesel generators, and RCIC function without manual assistance Normal reactor cooling systems should maintain adequate core temperatures, and prevent actuation of the Automatic Depressurization System; however selected relief valves may function to control pressure Turbine bypass system operates to maintain specified pressure value Steam system power-actuated pressure relief valves open and close at specified value Pressurizer spray valves open and close at specified values. Reactor coolant temperature/pressure relationship remains within prescribed values Pressurizer level is maintained within prescribed limits Steam generator level remains within prescribed limits	15.2.6 Loss of Nonemergency AC Power to the Station Auxiliaries