



OFFICE OF NUCLEAR REACTOR REGULATION

REVIEW STANDARD FOR
EXTENDED POWER UPRATES

APPROVED BY: /RA/

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Division of Licensing Project Management
Office of Nuclear Reactor Regulation

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DECEMBER 2003



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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	<ul style="list-style-type: none"> ● <i>Federal Register</i> ● Power Uprate Web site ● ADAMS 	None
12/2003	<p>Issuance of RS-001, Revision 0</p> <ul style="list-style-type: none"> ● 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 ● Moved the guidance for independent calculations from the individual matrices to Item (6) in Section 2.1 ● 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: <ul style="list-style-type: none"> • 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 ● Revised RS-001 to incorporate OGC comments, which included using the term "design bases" in lieu of "licensing bases" and modifying the regulatory evaluation sections of the template safety evaluations to incorporate additional wording from pertinent regulations such as the General Design Criteria. 	<ul style="list-style-type: none"> ● <i>Federal Register</i> ● Power Uprate Web site ● ADAMS 	Training sessions for staff in DLPM, DRIP, DIPM, DSSA & DE

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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 and template safety evaluation inserts in Section 3 of this review standard to identify any differences between the information in the review standard and the design 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 an NRC-approved topical report for an area covered by this review standard, the staff will review the application only to ensure that the licensee is applying the topical report under conditions for which the topical report was approved, using appropriate plant-specific inputs.

The staff will review plants against their design bases. Licensees are encouraged to provide, with their EPU applications, markups of the matrices in Section 2.1 and template safety evaluation inserts in Section 3 of this review standard to identify any differences between the information in the review standard and the design bases of their plants. This should help the staff identify areas where the criteria and/or guidance in the review standard does not apply to the plant under review. The staff does not intend to impose the criteria and/or guidance in this review standard on plants whose design bases do not include these criteria and/or guidance. No backfitting is intended or approved in connection with the issuance of this review standard.

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. EPU 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 the 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 necessary 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 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 lessons learned from the Maine Yankee experience (e.g., Report of the Maine Yankee Lessons Learned Task Group, dated December 1996), 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 the Maine Yankee experience and the feedback received from internal and external stakeholders.

Figure 1 provides a graphical representation of the development of the 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 (for which a hearing is not requested). (If a hearing is requested, the Project Manager will provide support to the Office of the General Counsel and arrange for staff to be available to support the hearing process.) 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

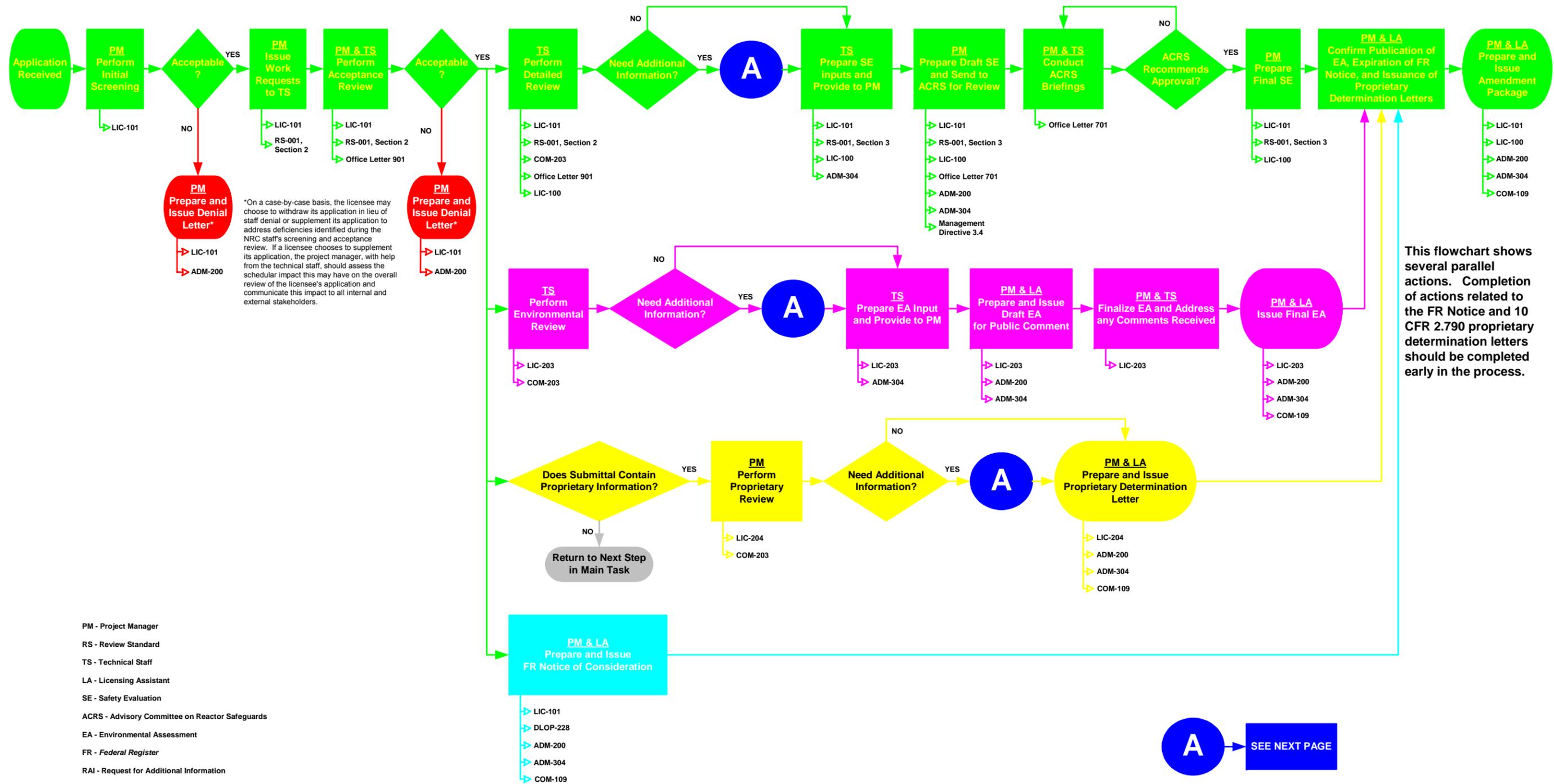
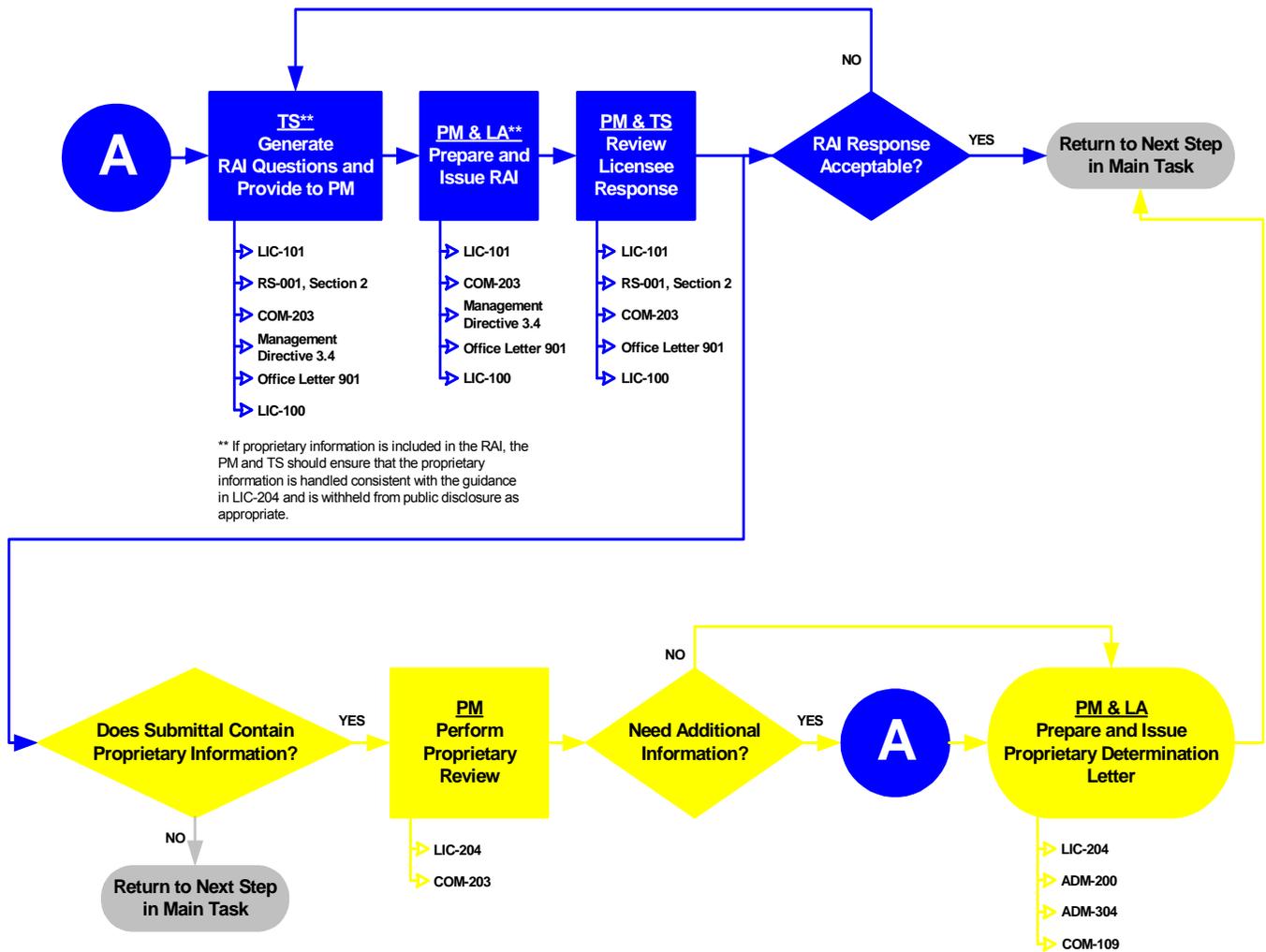


Figure 1.1-1 EPU Process Flow Chart

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 design 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 also encouraged to provide, with their EPU applications, markups of the matrices in Section 2.1 and template safety evaluation inserts in Section 3 of this review standard to identify any differences between the information in the review standard and the design bases of their plants. This should avoid potential delays and improve 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 design 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 design 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 design 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 also encouraged to provide, with their EPU applications, markups of the matrices in Section 2.1 and template safety evaluation inserts in Section 3 of this review standard to identify any differences between the information in the review standard and the design bases of their plants. This should avoid potential delays and improve 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 design 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 audits and/or independent calculations as deemed necessary and appropriate to support review of the licensee's application. In determining the need for performing audits and/or independent calculations, consider the following:
- confidence of the NRC staff in the models and/or methods used by the licensee
 - confidence of the NRC staff in the analysis results
 - familiarity of the NRC staff with the models and/or methods used by the licensee
 - prior use of the models and/or methods for similar plant designs and operating conditions and the NRC staff's experience related to such use
 - NRC staff experience with the impact of proposed changes on analysis results
 - available margin versus level of uncertainty in analysis results
 - efficiency gains that may result from performing audits and/or independent calculations

Any issues identified as a result of 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 EPU's	EMCB	SRXB	5.3.1 Draft Rev. 2 April 1996	GDC-14 GDC-31 10 CFR Part 50, App. H 10 CFR 50.60	RG 1.190	2.1.1	2.1.1	
Pressure-Temperature Limits and Upper-Shelf Energy	All EPU's	EMCB	SRXB	5.3.2 Draft Rev. 2 April 1996	GDC-14 GDC-31 10 CFR Part 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 EPU's	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 EPU's	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 EPU's	EMCB	EMEB SRXB	5.2.3 Draft Rev. 3 April 1996	GDC-1 10 CFR 50.55a GDC-4 GDC-31 10 CFR Part 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-31 10 CFR Part 50, App. G				
				5.3.3 Draft Rev. 2 April 1996					
				6.1.1 Draft Rev. 2 April 1996					
Leak-Before-Break	PWR EPU's	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 EPU's	EMCB		6.1.2 Draft Rev. 3 April 1996	10 CFR Part 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 EPU's	EMCB				Note 4*	2.1.6	2.1.8	
Steam Generator Tube Inservice Inspection	PWR EPU's	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 EPU's	EMCB		10.4.8 Draft Rev. 3 April 1996	GDC-14			2.1.10	
Chemical and Volume Control System (Including Boron Recovery System)	PWR EPU's	EMCB	SPLB SRXB	9.3.4 Draft Rev. 3 April 1996	GDC-14 GDC-29			2.1.11	
Reactor Water Cleanup System	BWR EPU's	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

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 EPU's	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 EPU's	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 EPU's	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 Part 100, App. A 10 CFR Part 50, App. B USI A-46		2.2.5	2.2.5	

Notes:

- As indicated in IN 2002-26 and Supplement 1 to IN 2002-26, the steam dryers and other plant components recently failed at Quad Cities Units 1 and 2 during operation under extended power uprate (EPU) conditions. The failures occurred as a result of high-cycle fatigue caused by increased flow-induced vibrations at EPU conditions. The staff's review of the reactor internals as part of EPU requests will cover detailed analyses of flow-induced vibration and acoustically-induced 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 EPU conditions. In addition, the staff is evaluating the need to address potential adverse effects on other plant components from the increased steam and feedwater flow under EPU conditions.

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

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 EPU's	EEIB		3.11 Draft Rev. 3 April 1996	10 CFR 50.49		2.3.1	2.3.1	
Offsite Power System	All EPU's	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 EPU's	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 EPU's	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 EPU's	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 relied upon 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
EPU = 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

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 EPU's	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		2.4.1	2.4.1	
Engineered Safety Features Systems	All EPU's	EEIB		7.3 Rev. 4 June 1997	GDC-20 GDC-21 GDC-22 GDC-23 GDC-24		2.4.1	2.4.1	
Safety Shutdown Systems	All EPU's	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 EPU's	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 EPU's	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 EPU's	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
EPU = extended power uprates
GDC = General Design Criterion
I&C = instrumentation and controls
PWR = pressurized-water reactor
SRP = Standard Review Plan

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	EPU 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	EPU 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	EPU 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)	EPU 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)	EPU 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 EPU's 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	EPU's 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 EPU's except where the application demonstrates that previous analysis is bounding	SPLB		9.5.1 Rev. 3 July 1981	10 CFR 50.48 10 CFR Part 50, App. R GDC-3 GDC-5	Note 1*	2.5.1.4	2.5.1.4	
Pressurizer Relief Tank	PWR EPU's 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 EPU's 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	EPU's 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	EPU's 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 EPU's 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 EPU's 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 EPU's 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 EPU's 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 EPU's 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 EPU's 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 EPU's 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 EPU's 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 EPU's 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	EPU's 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 Part 50, App. I		2.5.5.1	2.5.6.1	
Liquid Waste Management Systems	EPU's 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 Part 50, App. I		2.5.5.2	2.5.6.2	
Solid Waste Management Systems	EPU's 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 Part 71		2.5.5.3	2.5.6.3	
Emergency Diesel Engine Fuel Oil Storage and Transfer System	EPU's 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	

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	
Light Load Handling System (Related to Refueling)	EPU's 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 1 to this matrix.
2. Supplemental guidance for review of spent fuel pool cooling is provided in Attachment 2 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
EPU = 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 = Probabilistic Safety Assessment Branch
SRP = Standard Review Plan

ATTACHMENT 1 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 necessary. 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 integrity of the reactor pressure vessel 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 is sufficient to achieve and maintain 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 flow rates needed 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 2 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 generally consistent with GDC-61. However, later guidance contained in Section 9.1.3 of the Standard Review Plan applied GDC-44 to the SFP cooling system. GDC-44 requires, in part, that a licensee provide a cooling system that is capable of accomplishing its safety function with or without offsite sources of power, assuming a single failure. To satisfy these criteria, each licensee should demonstrate that there is adequate SFP cooling capacity and should also demonstrate 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 may be accomplished only after 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) to ensure that the licensee has demonstrated the adequacy of the SFP cooling system.

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 design basis of the facility; and
- (4) the availability of two reliable sources of makeup water, one having sufficient capacity to make up for evaporation following a total loss of forced cooling. Only one source need have this capacity because the heat load and boil-off rate decrease rapidly with time from the peak value such that a much lower makeup rate would be effective in extending the recovery time.

The reliability of the systems relied upon to meet these guidelines should be maintained consistent with the plant's current design 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) should 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 should 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) should 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 should remain below the design temperature of the SFP structure and liner, and calculated peak storage cell temperature should 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 should 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 should have a capacity that is equal to or greater than the calculated boil-off rate so that the SFP level can be maintained. Only one source need have this capacity because the heat load and boil-off rate decrease rapidly with time from the peak value such that a much lower makeup rate would be effective in extending the recovery time.

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	EPU 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	GDC-13 GDC-16 GDC-38 GDC-50 GDC-64			2.6.1	
				6.2.1.1.A Rev. 2 July 1981					
Ice Condenser Containments	EPU 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	GDC-13 GDC-16 GDC-38 GDC-50 GDC-64			2.6.1	
				6.2.1.1.B Rev. 2 July 1981					

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	EPU 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	GDC-4 GDC-13 GDC-16 GDC-50 GDC-64		2.6.1		
				6.2.1.1.C Rev. 6 Aug. 1984					
Subcompartment Analysis	All EPUs except where the application demonstrates that previous analysis is bounding	SPSB		6.2.1 Rev. 2 July 1981	GDC-4 GDC-50		2.6.2	2.6.2	
				6.2.1.2 Rev. 2 July 1981					
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	GDC-50 10 CFR Part 50, App. K		2.6.3.1	2.6.3.1	
				6.2.1.3 Rev. 1 July 1981					
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	GDC-50			2.6.3.2	
				6.2.1.4 Rev. 1 July 1981					

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	EPU's 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 EPU's 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	EPU's 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 EPU's except where the application demonstrates that previous analysis is bounding	SPSB	SRXB	6.2.1 Rev. 2 July 1981	10 CFR 50.46 10 CFR Part 50, App. K			2.6.6	
				6.2.1.5 Rev. 2 July 1981					

LIST OF ACRONYMS FOR MATRIX 6

BWR = boiling-water reactor

CFR = *Code of Federal Regulations*

DG = draft guide

EPU = extended power uprates

GDC = General Design Criterion

PWR = pressurized-water reactor

SPSB = Probabalistic Safety Assessment Branch

SRP = Standard Review Plan

SRXB = Reactor Systems Branch

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 EPU's 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 EPU's 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 EPU's 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 EPU's 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 EPU's 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 EPU's 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 EPU's 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:

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

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 EPU's	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 EPU's	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 EPU's	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		Acceptance Review Checklist
							BWR	PWR	
Functional Design of Control Rod Drive System	All EPU's	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 EPU's	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 EPU's	SRXB		5.2.2 Draft Rev. 3 April 1996	GDC-15 GDC-31			2.8.4.3	
Reactor Core Isolation Cooling System	BWR EPU's	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 EPU's	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	

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	
Emergency Core Cooling System	All EPU's	SRXB		6.3 Draft Rev. 3 April 1996	GDC-4 GDC-27 GDC-35 10 CFR 50.46 10 CFR Part 50, App. K	Note 6*	2.8.5.6.2	2.8.5.6.3	
Standby Liquid Control System	BWR EPU's	SRXB	EMCB SPLB	9.3.5 Draft Rev. 3 April 1996	GDC-26 GDC-27 10 CFR 50.62(c)(4)	Note 10*	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 EPU's	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 EPU's	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 EPU's	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 EPU's	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	

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 Normal Feedwater Flow	All EPU's	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 EPU's	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	
Loss of Forced Reactor Coolant Flow Including Trip of Pump Motor and Flow Controller Malfunctions	All EPU's	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 EPU's	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 EPU's	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 EPU's	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 EPU's	SRXB		15.4.3 Draft Rev. 3 April 1996	GDC-10 GDC-20 GDC-25	Note 7*		2.8.5.4.3	

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	
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 EPU's	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 EPU's	SRXB		15.4.6 Draft Rev. 2 April 1996	GDC-10 GDC-15 GDC-26	Note 7*		2.8.5.4.5	
Spectrum of Rod Ejection Accidents	PWR EPU's	SRXB		15.4.8 Draft Rev. 3 April 1996	GDC-28	Note 7*		2.8.5.4.6	
Spectrum of Rod Drop Accidents	BWR EPU's	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 EPU's	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 EPU's	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 EPU's	SRXB		15.6.3 Draft Rev. 3 April 1996	Note 7*	Note 7*		2.8.5.6.2	

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	
Loss-of Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary	All EPU's	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 EPU's	SRXB				Note 7* Note 10*	2.8.5.7	2.8.5.7	
New Fuel Storage	EPU applications that request approval for new fuel design.	SRXB		9.1.1 Draft Rev. 3 April 1996	GDC-62		2.8.6.1	2.8.6.1	
Spent Fuel Storage	EPU applications that request approval for new fuel design.	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) need 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 future fuel burnup limits. 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 determination of 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 design-basis case and the upper-bound case and any impact of the changes in PCTs on the use of the design methods for the power uprate.
7. The review:
 - confirms that 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.
 - confirms that all changes of reactor protection system trip delays are correctly addressed and accounted for in the analyses.
 - (for PWRs) confirms that steam generator plugging and asymmetry limits are accounted for in the analyses.
 - (for PWRs) covers 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 identified problems 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 increase 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 EPU.
 - 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 = boiling-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

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 EPU's	SPSB		11.1 Draft Rev. 3 April 1996	10 CFR Part 20 10 CFR Part 50, App. I GDC-60		2.9.1	2.9.1	
Radiological Consequence Analyses Using Alternative Source Terms	EPU's 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 Part 51 10 CFR Part 50, App. E NUREG-0737		2.9.2	2.9.2	
Radiological Consequences of Main Steamline Failures Outside Containment for a PWR	PWR EPU's 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 Part 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*			

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 Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break	EPU's 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 Part 100	Notes 5, 8, 9, 27*		2.9.3	
				6.4 Draft Rev. 3 April 1996	GDC-19	Notes 1, 2, 3, 28, 29*			
Radiological Consequences of a Control Rod Ejection Accident	PWR EPU's 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 Part 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 EPU's 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 Part 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	EPU's 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 Part 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 EUs 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 Part 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 EUs 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 Part 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	EUs that do not utilize alternative source term	SPSB	SPLB	15.6.5, App. A Draft Rev. 2 April 1996	10 CFR Part 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	EPU's that do not utilize alternative source term	SPSB	SPLB	15.6.5, App. B Draft Rev. 2 April 1996	10 CFR Part 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 EPU's that do not utilize alternative source term	SPSB		15.6.5, App. D Draft Rev. 2 April 1996	10 CFR Part 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	EPU's that do not utilize alternative source term	SPSB	SPLB	15.7.4 Draft Rev. 2 April 1996	10 CFR Part 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	EPU's that do not utilize alternative source term	SPSB	EMEB SPLB	15.7.5 Draft Rev. 3 April 1996	10 CFR Part 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," as it applies to irradiated fuel, to the applicability section of certain technical specifications. The proposed change is 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 it applies to irradiated fuel, 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

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 EPU's	IEPB		12.2 Draft Rev. 3 April 1996	10 CFR Part 20		2.10.1	2.10.1	
Radiation Protection Design Features	All EPU's	IEPB		12.3-4 Draft Rev. 3 April 1996	10 CFR Part 20 GDC-19	Note 1*	2.10.1	2.10.1	
Operational Radiation Protection Program	All EPU's	IEPB		12.5 Draft Rev. 3 April 1996	10 CFR Part 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*

EPU = extended power uprates

GDC = General Design Criterion

IEPB = Emergency Preparedness and Plant Support Branch

PWR = pressurized-water reactor

SRP = Standard Review Plan

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 EPU's	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 EPU's	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 EPU's	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	

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	
Human Factors Engineering	All EPUs	IROB		18.0** Draft Rev. 0 April 1996	Specific review questions are provided in the template safety evaluations.		2.11	2.11	

*The staff is currently finalizing SRP Sections 13.2.1, 13.2.2, and 13.5.2.1. While these SRP Sections are being finalized, the staff will continue to use the versions issued in December 2002 for interim use and public comment. Once finalized, the staff will use the new versions of these SRP Sections.

**The staff received significant comment on draft SRP Chapter 18.0 that was issued in December 2002 for interim use and public comment. The staff is working on finalizing this SRP. However, due to the significance of the comments received, the staff will use Draft SRP Chapter 18.0, Revision 0, dated April 1996.

LIST OF ACRONYMS FOR MATRIX 11

BWR = boiling-water reactor

EPU = 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

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 EPU's	IEPB	EEIB EMCB EMEB IROB SPLB SPSB SRXB	14.2.1* Draft Rev. 0 Dec. 2002	Entire Section		2.12	2.12	

*The staff is currently finalizing SRP Section 14.2.1. While this SRP Section is being finalized, the staff will continue to use the version issued for interim use and public comment in December 2002. Once finalized, the staff will use the new version.

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
EPU = extended power uprates
IEPB = Emergency Preparedness and Plant Support Branch
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

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 EPU's	SPSB				Note 1* RG 1.174 RIS 2001-02	2.13	2.13	

Notes:

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

LIST OF ACRONYMS FOR MATRIX 13

BWR = boiling-water reactor

EPU = 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

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 necessary, 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 are 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 risk, some increase in risk is expected for larger power uprates. Depending on the type of plant-specific modifications necessary 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 original licensed thermal power.

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 in the regulations. Such issues could represent the "special circumstances" that would call for a more detailed risk review to determine the acceptability of the extended power uprate license application. These reviews can involve 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 needed for the staff to be able to determine the acceptability of the license application. This may result in the licensee and/or staff obtaining 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 needs 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 are 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 necessitate 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 reflect 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, 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 an extended power uprate. 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 licensees' 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 for these external events associated with implementing the extended power uprate 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 points for determining if the potential for special circumstances exists are 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., Δ CDF and Δ LERF). 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 are 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 reflect 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 2 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 3 to Matrix 13 of RS-001.

The results of these simplistic approaches should not be used as the sole basis for determining the acceptability 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 warrant 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 evaluating this risk information to determine if there is reasonable assurance of adequate protection. If NRC management agrees with the staff that special circumstances appear 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 contains enough information for the reviewer to evaluate the application in accordance with 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 need 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 respect to 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 called for in response to 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 2 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 a 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 (as documented in meeting summaries, notes to file, etc.), 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 (as documented in meeting summaries, notes to file, etc.), such breaches would be unanticipated and/or inadvertent. The small increase in decay heat due to the proposed extended power

update (EPU) will reduce the time available for operator actions, such as to achieve containment closure. However, even for the containment breach that takes the longest to close (i.e., the equipment hatch), the licensee has demonstrated that such a breach can be closed within 5 minutes to 15 minutes, and the estimated time to boiling would be greater than 18 minutes for EPU conditions. For the pre-EPU conditions, the time to boil is estimated at over 20 minutes. Therefore, the operator's ability to inject before core damage and close containment before boiling should not be significantly changed, since (1) there is margin between the longest time needed to close containment 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 3 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. Therefore, as set forth in that SER, the staff concluded 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.

As a consequence of using the EPRI SMA methodology, the licensee 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.

Although the IPEEE indicates that it is a 0.3g focused-scope SMA, this scenario involves equipment with a 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 $1\text{E-}5/\text{year}$, with a change in risk due to the uprate of about $1\text{E-}8/\text{year}$.

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 $1.7\text{E-}5/\text{year}$. The seismic risk associated with the remainder of the plant having a HCLPF at 0.3g PGA using the same approach is about $3.1\text{E-}6/\text{year}$. Thus, based on the staff’s approximation, the total seismic CDF is estimated to be about $2\text{E-}5/\text{year}$.

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 design 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 design basis of the plant. [Note: This is most likely to occur with respect to the General Design Criteria (10 CFR Part 50, Appendix A), which may need to be replaced by plant-specific principal design criteria (PDC). The PDC are usually based on proposed draft GDC.] 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. If a "Conclusion" section summarizes more than one technical evaluation, include an intermediate conclusion in each technical evaluation (e.g., see Section 2.2.2 of Insert 2 for RS-001 Section 3.2 - BWR Template Safety Evaluation).
- (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.
- (11) Modify, as necessary, the acronym list that is attached to the template safety evaluation to ensure that it accurately reflects the acronyms defined in the plant-specific safety evaluation.

It may be necessary to modify the license to include license conditions to capture certain future licensee actions discussed in the EPU application. These actions are typically included as commitments in the EPU application and may include things such as plant modifications, analyses, and updates to licensee-controlled documents. In addition, in cases where a licensee proposes to implement the EPU in multiple stages, it may be appropriate to modify the license to include license conditions to limit plant operation to lower than the full EPU power level pending completion of certain actions. To determine if such actions are appropriate for inclusion in the license as license conditions, refer to the guidance in NRR Office Instruction LIC-101, "License Amendment Review Procedures."

For EPUs to be implemented in one stage, the PM should consider including conditions in the implementation section of the amendment to appropriately capture near-term licensee actions meeting the threshold for inclusion in the license as license conditions.

For EPUs to be implemented in multiple stages, the PM should consider including conditions in the license to appropriately capture longer-term licensee actions meeting the threshold for inclusion in the license as license conditions. Including these actions in the license is appropriate due to the licensee's extended schedule for implementing the EPU.

SECTION 3.2 of RS-001

TEMPLATE SAFETY EVALUATION

for

**BOILING-WATER REACTOR
EXTENDED POWER UPRATE**

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Attachment: List of Acronyms

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 design basis of the plant. Where differences exist between the plant-specific design basis and RS-001, the licensee described the differences and provided evaluations consistent with the design 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 necessary to implement 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 design-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 the affects of the 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

As described above, 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's 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 necessary to implement the proposed EPU, and new conditions of operation necessary 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. *If no comments were received, use the following sentence: [No comments were received on the draft Environmental Assessment.] If comments were received, use the following sentence: [The NRC staff received comments which were addressed in the final environmental assessment.]* 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, Revision 0, "Review Standard for Extended Power Uprates," December 2003.
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	<i>Code of Federal Regulations</i>
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 (or criteria)
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 focused 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, insofar as it requires that the reactor coolant pressure boundary (RCPB) be designed, fabricated, erected, and tested so as to have an extremely low probability of rapidly propagating fracture; (2) GDC-31, insofar as it requires that the RCPB be designed with margin sufficient to assure that, under specified conditions, it will behave in a nonbrittle manner and the probability of a rapidly propagating fracture is minimized; (3) 10 CFR Part 50, Appendix H, which provides for monitoring changes in the fracture toughness properties of materials in the reactor vessel beltline region; and (4) 10 CFR 50.60, which requires 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 10 CFR Part 50, Appendix H, and 10 CFR 50.60, and will provide the licensee with information to ensure continued compliance with GDC-14 and GDC-31 in this respect 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 covered the P-T limits methodology and the calculations for the number of effective full power years specified for the proposed EPU, 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, insofar as it requires that the RCPB be designed, fabricated, erected, and tested so as to have an extremely low probability of rapidly propagating fracture; (2) GDC-31, insofar as it requires that the RCPB be designed with margin sufficient to assure that, under specified conditions, it will behave in a nonbrittle manner and the probability of a rapidly propagating fracture is minimized; (3) 10 CFR Part 50, Appendix G, which specifies fracture toughness requirements for ferritic components of the RCPB; and (4) 10 CFR 50.60, which requires 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 operation under the proposed EPU conditions. Based on this, the NRC staff concludes that the proposed P-T limits will continue to meet the requirements of 10 CFR Part 50, Appendix G, and 10 CFR 50.60 and will enable the licensee to comply with GDC-14 and GDC-31 in this respect 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 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 covered 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 with respect to material specifications, welding controls, and inspection 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 covered 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) 10 CFR 50.55a and GDC-1, insofar as they require that SSCs important to safety be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety functions to be performed; (2) GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents; (3) GDC-14, insofar as it requires that the RCPB be designed, fabricated, erected, and tested so as to have an extremely low probability of rapidly propagating fracture; (4) GDC-31, insofar as it requires that the RCPB be designed with margin sufficient to assure that, under specified conditions, it will behave in a nonbrittle manner and the probability of a rapidly propagating fracture is minimized; and (5) 10 CFR Part 50, Appendix G, which specifies fracture toughness requirements for ferritic components 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 covered protective coating systems used inside the containment for their suitability for and stability under design-basis loss-of-coolant accident (DBLOCA) 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, which states quality assurance requirements for the design, fabrication, and construction of safety-related SSCs and (2) Regulatory Guide 1.54, Revision 1, for guidance on 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 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. 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 due to FAC depend on velocity of flow, fluid 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 has reviewed 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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. The NRC staff further 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 a path for removal of reactor coolant when necessary. Portions of the RWCS comprise the RCPB. The NRC staff's review of the RWCS included 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 consisted of evaluating the adequacy of the plant's TSs in these areas under the proposed EPU conditions. The NRC's acceptance criteria for the RWCS are based on (1) GDC-14, insofar as it requires that the RCPB be designed, fabricated, erected, and tested so as to have an extremely low probability of rapidly propagating fracture; (2) GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents; and (3) GDC-61, insofar as it requires that systems that contain radioactivity be designed with appropriate confinement. Specific review criteria are contained in SRP Section 5.4.8.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 conducted 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 covered (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 (ISI) programs or the use of special protective devices such as pipe-whip restraints, (3) 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 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, which requires SSCs important to safety to be 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 has reviewed 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 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 covered (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 included 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, insofar as they require that SSCs important to safety be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety functions to be performed; (2) GDC-2, insofar as it requires that SSCs important to safety be designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions; (3) GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents; (4) GDC-14, insofar as it requires that the RCPB be designed, fabricated, erected, and tested so as to have an extremely low probability of rapidly propagating fracture; and (5) GDC-15, insofar as it requires that the RCS be designed with margin sufficient to ensure that the design conditions of the RCPB are not exceeded during any condition of normal operation. 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 (NSSS) piping, components, and supports. Include an intermediate conclusion in the form of "Because [summarize reasons], the NSSS piping, components, and supports are adequate under the proposed EPU conditions."]

Balance-of-Plant Piping, Components, and Supports

[Insert technical evaluation for balance-of-plant piping, components, and supports. Include an intermediate conclusion in the form of "Because [summarize reasons], the balance-of-plant piping, components, and supports are adequate under the proposed EPU conditions."]

Reactor Vessel and Supports

[Insert technical evaluation for reactor vessel and supports. Include an intermediate conclusion in the form of "Because [summarize reasons], the reactor vessel and supports are adequate under the proposed EPU conditions."]

Control Rod Drive Mechanism

[Insert technical evaluation for control rod drive mechanism. Include an intermediate conclusion in the form of “Because [summarize reasons], the control rod drive mechanism is adequate under the proposed EPU conditions.”]

Recirculation Pumps and Supports

[Insert technical evaluation for reactor coolant pumps and supports. Include an intermediate conclusion in the form of “Because [summarize reasons], the recirculation pumps and supports are adequate under the proposed EPU conditions.”]

Conclusion

The NRC staff has reviewed the licensee’s evaluations related to the structural integrity of pressure-retaining components and their supports. For the reasons set forth above, the NRC staff concludes that the licensee has adequately addressed the effects of the proposed EPU on these components and their supports. Based on the above, 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 reviewed 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 loss-of-coolant accidents (LOCAs), and the identification of design transient occurrences. The NRC staff's review covered (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 included a comparison of the resulting stresses and CUFs against the corresponding Code-allowable limits. The NRC's acceptance criteria are based on (1) 10 CFR 50.55a and GDC-1, insofar as they require that SSCs important to safety be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety functions to be performed; (2) GDC-2, insofar as it requires that SSCs important to safety be designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions; (3) GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents; and (4) GDC-10, insofar as it requires that the reactor core be designed 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 the reactor internals and core supports. 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 included 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 focused on the effects of the proposed EPU on the required functional performance of the valves and pumps. The review also covered 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 evaluated 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, insofar as it requires that SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed; (2) GDC-37, GDC-40, GDC-43, and GDC-46, insofar as they require that the emergency core cooling system (ECCS), the containment heat removal system, the containment atmospheric cleanup systems, and the cooling water system, respectively, be designed to permit appropriate periodic testing to ensure the leak-tight integrity and performance of their active components; (3) GDC-54, insofar as it requires that piping systems penetrating containment be designed with the capability to periodically test the operability of the isolation valves to determine if valve leakage is within acceptable limits; and (4) 10 CFR 50.55a(f), insofar as it requires that pumps and valves subject to that section must meet the inservice testing program requirements identified in that section. 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessments 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 safety-related pumps and valves. 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 to preventing significant releases of radioactive materials to the environment are also covered by this section. The NRC staff's review focused 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, insofar as it requires that SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed; (2) GDC-30, insofar as it requires that components that are part of the RCPB be designed, fabricated, erected, and tested to the highest quality standards practical; (3) GDC-2, insofar as it requires that SSCs important to safety be designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions; (4) 10 CFR Part 100, Appendix A, which sets forth the principal seismic and geologic considerations for the evaluation of the suitability of plant design bases established in consideration of the seismic and geologic characteristics of the plant site; (5) GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents; (6) GDC-14, insofar as it requires that the RCPB be designed, fabricated, erected, and tested so as to have an extremely low probability of rapidly propagating fracture; and (7) 10 CFR Part 50, Appendix B, which sets quality assurance requirements for safety-related equipment. Specific review criteria are contained in SRP Section 3.10.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 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 was 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, which sets forth requirements for 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 electrical equipment. The NRC staff further concludes that the electrical equipment will continue to meet 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 covered 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 focused on whether the loss of the nuclear unit, the largest operating unit on the grid, or the most critical transmission line will result in the loss of offsite power (LOOP) 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 alternating current (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 covered 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, insofar as it requires the system to have the capacity and capability to perform its intended functions during anticipated operational occurrences and accident conditions. Specific review criteria are contained in SRP Sections 8.1 and 8.3.1.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 direct current (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 covered 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, insofar as it requires the system to have the capacity and capability to perform its intended functions during anticipated operational occurrences and accident conditions. Specific review criteria are contained in SRP Sections 8.1 and 8.3.2

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 LOOP 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 focused on the impact of the proposed EPU on the plant's ability to cope with and recover from an 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 an 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 (including 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 conducted 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 necessary for the proposed EPU are adequately designed such that the systems continue to meet their safety functions. The NRC staff's review was 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 necessary to achieve the proposed EPU are consistent with the plant's design 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 conducted a review in the area of flood protection to ensure that SSCs important to safety are protected from flooding. The NRC staff's review covered flooding of SSCs important to safety from internal sources, such as those caused by failures of tanks and vessels. The NRC staff's review focused 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 an uncontaminated drainage system. The NRC staff's review of the EFDS included the collection and disposal of liquid effluents outside containment.

The NRC staff's review focused on any changes in fluid volumes or pump capacities that are necessary 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 insofar as they require the EFDS to be designed 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 focused on changes in flooding analyses that are necessary due to increases in fluid volumes or installation of larger capacity pumps or piping needed 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 covered pressurized components and systems, and high-speed rotating machinery. The NRC staff's review was conducted to ensure that safety-related SSCs are adequately protected from internally generated missiles. In addition, for cases where safety-related SSCs are located in areas containing non-safety-related SSCs, the NRC staff reviewed 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 focused 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 focused 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, and relates to 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 conducted 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 included high and moderate energy fluid system piping located outside of containment. The NRC staff's review focused on the effects of pipe failures on plant 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, which requires, in part, that SSCs important to safety be 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the changes that are necessary 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 the dynamic effects of 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 focused 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 will 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, insofar as they require the development of an FPP to ensure, among other things, the capability to safely shut down the plant; (2) GDC-3, insofar as it requires that (a) SSCs important to safety be designed and located to minimize the probability and effect of fires, (b) noncombustible and heat resistant materials be used, and (c) fire detection and fighting systems be provided and designed to minimize the adverse effects of fires on SSCs important to safety; (3) GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions. Specific review criteria are contained in SRP Section 9.5.1, as supplemented by the guidance provided in Attachment 2 to Matrix 5 of Section 2.1 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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, Appendix R to 10 CFR Part 50, 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 covered 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 focused 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, insofar as it requires that the containment atmosphere cleanup system be provided to reduce the concentration of fission products released to the environment following postulated accidents. Specific review criteria are contained in SRP Section 6.5.3.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 in fission products and changes in expected environmental conditions that would result from the proposed EPU. 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 focused 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, insofar as it requires that the plant design include means to control the release of radioactive effluents; and (2) GDC-64, insofar as it requires that means be provided for monitoring effluent discharge paths and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences and postulated accidents. Specific review criteria are contained in SRP Section 10.4.2.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 reviewed 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, insofar as it requires that the plant design include means to control the release of radioactive effluents; and (2) GDC-64, insofar as it requires that means be provided for monitoring effluent discharge paths and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences and postulated accidents. Specific review criteria are contained in SRP Section 10.4.3.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 focused 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, insofar as it requires that piping systems penetrating containment be provided with leakage detection and isolation capabilities. Specific review criteria are contained in SRP Section 6.7.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 focused 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, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions, (2) GDC-44, insofar as it requires that a system with the capability to transfer heat loads from safety-related SSCs to a heat sink under both normal operating and accident conditions be provided, and (3) GDC-61, insofar as it requires that fuel storage systems be designed with RHR capability reflecting the importance to safety of decay heat removal, 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 1 to Matrix 5 of Section 2.1 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 covered the characteristics of the station SWS components with respect to their functional performance as affected by adverse operational (i.e., water hammer) conditions, abnormal operational conditions, and accident conditions (e.g., a LOCA with the LOOP). The NRC staff's review focused on the additional heat load that would result from the proposed EPU. The NRC's acceptance criteria are based on (1) GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, including flow instabilities and loads (e.g., water hammer), maintenance, testing, and postulated accidents; (2) GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions; and (3) GDC-44, insofar as it requires that a system with the capability to transfer heat loads from safety-related SSCs to a heat sink under both normal operating and accident conditions be provided. 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 on system performance that would result from the proposed EPU. 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 covered reactor auxiliary cooling water systems that are required for (1) safe shutdown during normal operations, anticipated operational occurrences, and 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 covered 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 was placed on the cooling water systems for safety-related components (e.g., ECCS equipment, ventilation equipment, and reactor shutdown equipment). The NRC staff's review focused on the additional heat load that would result from the proposed EPU. The NRC's acceptance criteria for the reactor auxiliary cooling water system are based on (1) GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation including flow instabilities and attendant loads (i.e., water hammer), maintenance, testing, and postulated accidents; (2) GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions; and (3) GDC-44, insofar as it requires that a system with the capability to transfer heat loads from safety-related SSCs to a heat sink under both normal operating and accident conditions be provided. 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 focused on the impact that the proposed EPU has on the decay heat removal capability of the UHS. Additionally, the NRC staff's review included 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, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety; and (2) GDC-44, insofar as it requires that a system with the capability to transfer heat loads from safety-related SSCs to a heat sink under both normal operating and accident conditions be provided. Specific review criteria are contained in SRP Section 9.2.5.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 following implementation of the proposed EPU. 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 NSSS to the power conversion system and various safety-related and non-safety-related auxiliaries. The NRC staff's review focused 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, insofar as it requires that SSCs important to safety be protected against dynamic effects, including the effects missiles, pipe whip, and jet impingement forces associated with pipe breaks; and (2) GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions. Specific review criteria are contained in SRP Section 10.3.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 5. 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 (TBS). 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 focused 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 TBS. The NRC's acceptance criteria for the MC system are based on GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents. Specific review criteria are contained in SRP Section 10.4.1.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 TBS and thereby continue to meet GDC-60 with respect to controlling releases of radioactive effluents. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the MC system.

2.5.4.3 Turbine Bypass

Regulatory Evaluation

The 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 MSIV leakage control system, 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 focused 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, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents (including pipe breaks or malfunctions of the TBS), and (2) GDC-34, insofar as it requires that a RHR system be provided to transfer fission product decay heat and other residual heat from the reactor core at a rate such that SAFDLs and the design conditions of the RCPB are not exceeded. Specific review criteria are contained in SRP Section 10.4.4.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 a particular temperature, pressure, and flow rate to the reactor. The only part of the CFS classified as safety-related is the feedwater piping from the NSSS up to and including the outermost containment isolation valve. The NRC staff's review focused 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, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation including possible fluid flow instabilities (e.g., water hammer), maintenance, testing, and postulated accidents; (2) GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions; and (3) GDC-44, insofar as it requires that a system with the capability to transfer heat loads from safety-related SSCs to a heat sink under both normal operating and accident conditions be provided, and that the system be provided with suitable isolation capabilities to assure the safety function can be accomplished with electric power available from only the onsite system or only the offsite system, assuming a single failure. Specific review criteria are contained in SRP Section 10.4.7.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 the condenser air removal system; the gland seal exhaust and the mechanical vacuum pump operation exhaust; and the building ventilation system exhausts. The NRC staff's review focused on the effects that the proposed EPU may have on (1) the design criteria of the gaseous waste management systems, (2) methods of treatment, (3) expected releases, (4) principal parameters used in calculating the releases of radioactive materials in gaseous effluents, and (5) design features for precluding 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, insofar as it provides for demonstrating that annual average concentrations of radioactive materials released at the boundary of the unrestricted area do not exceed specified values; (2) GDC-3, insofar as it requires that (a) SSCs important to safety be designed and located to minimize the probability and effect of fires, (b) noncombustible and heat resistant materials be used, and (c) fire detection and fighting systems be provided and designed to minimize the adverse effects of fires on SSCs important to safety; (3) GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents; (4) GDC-61, insofar as it requires that systems that contain radioactivity be designed with appropriate confinement; and (5) 10 CFR Part 50, Appendix I, Sections II.B, II.C, and II.D, which set 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 focused on the effects that the proposed EPU may 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, insofar as it provides for demonstrating that annual average concentrations of radioactive materials released at the boundary of the unrestricted area do not exceed specified values; (2) GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents; (3) GDC-61, insofar as it requires that systems that contain radioactivity be designed with appropriate confinement; and (4) 10 CFR Part 50, Appendix I, Sections II.A and II.D, which set 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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) focused on the effects that the proposed EPU may 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, insofar as it provides for demonstrating that annual average concentrations of radioactive materials released at the boundary of the unrestricted area do not exceed specified values; (2) GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents; (3) GDC-63, insofar as it requires that systems be provided in waste handling areas to detect conditions that may result in excessive radiation levels, (4) GDC-64, insofar as it requires that means be provided for monitoring effluent discharge paths and the plant environs for radioactivity that may be released from normal operations, including AOOs, and postulated accidents; and (5) 10 CFR Part 71, which states requirements for radioactive material packaging. Specific review criteria are contained in SRP Section 11.4.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 supplies of sufficient capacity to perform their safety functions (e.g., power diesel engine-driven generator sets), assuming a single failure. The NRC staff's review focused on increases in emergency diesel generator electrical demand and the resulting increase in the amount of fuel oil necessary for the system to perform its safety function. The NRC's acceptance criteria for the emergency diesel engine fuel oil storage and transfer system are based on (1) GDC-4, insofar as it requires that SSCs important to safety be protected against dynamic effects, including missiles, pipe whip, and jet impingement forces associated with pipe breaks; (2) GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions; and (3) GDC-17, insofar as it requires onsite power supplies to have sufficient independence and redundancy to perform their safety functions, assuming a single failure. Specific review criteria are contained in SRP Section 9.5.4.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 and the loading of spent fuel into shipping casks. The NRC staff's review covered the avoidance of criticality accidents, radioactivity releases resulting from damage to irradiated fuel, and unacceptable personnel radiation exposures. The NRC staff's review 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, insofar as it requires that systems that contain radioactivity be designed with appropriate confinement and with suitable shielding for radiation protection; and (2) GDC-62, insofar as it requires that criticality be prevented. Specific review criteria are contained in SRP Section 9.1.4.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 NRC staff's review for the primary containment functional design covered (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 analytical models used for containment analysis. The NRC's acceptance criteria for the primary containment functional design are based on (1) GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, and that such SSCs be protected against dynamic effects; (2) GDC-16, insofar as it requires that reactor containment be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment; (3) GDC-50, insofar as it requires that the containment and its associated heat removal systems be designed so that the containment structure can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated temperature and pressure conditions resulting from any LOCA; (4) GDC-13, insofar as it requires that instrumentation be provided to monitor variables and systems over their anticipated ranges for normal operation and for accident conditions, as appropriate, to assure adequate safety; and (5) GDC-64, insofar as it requires that means be provided to monitor 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 covered the determination of the design differential pressure values for containment subcompartments. The NRC staff's review focused on the effects of the increase in mass and energy release into the containment due to operation at EPU conditions, and the resulting increase in pressurization. The NRC's acceptance criteria for subcompartment analyses are based on (1) GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, and that such SSCs be protected against dynamic effects, and (2) GDC-50, insofar as it requires that containment subcompartments be designed with sufficient margin to prevent fracture of the structure due to the calculated pressure differential conditions across the walls of the subcompartments. Specific review criteria are contained in SRP Section 6.2.1.2.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 containment SSCs important to safety will continue to be protected from the dynamic effects resulting from 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, 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 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 covered 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, insofar as it requires that sufficient conservatism be provided in the mass and energy release analysis to assure that containment design margin is maintained and (2) 10 CFR Part 50, Appendix K, insofar as it identifies sources of energy during a LOCA. Specific review criteria are contained in SRP Section 6.2.1.3.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 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 reactions 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 covered (1) the production and accumulation of 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 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, insofar as it requires that plants be provided with the capability for controlling combustible gas concentrations in the containment atmosphere; (2) GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions; (3) GDC-41, insofar as it requires that systems be 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; (4) GDC-42, insofar as it requires that systems required by GDC-41 be designed to permit appropriate periodic inspection; and (5) GDC-43, insofar as it requires that systems required by GDC-41 be designed to permit appropriate periodic testing. **[Include the following sentence for BWRs with Mark III containments: Additional requirements based on 10 CFR 50.44 for control of combustible gas apply to plants with a Mark III type of containment that do not rely on an inerted atmosphere to control hydrogen inside the containment.]** Specific review criteria are contained in SRP Section 6.2.5.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 and GDCs 5, 41, 42, and 43 as discussed 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 focused 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, insofar as it requires that a containment heat removal system be provided, and that its function shall be to rapidly reduce the containment pressure and temperature following a LOCA and maintain 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 with respect to 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 covered (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 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, insofar as it requires that SSCs important to safety be designed to accommodate the effects of environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, and be 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, insofar as it requires that reactor containment and associated systems be 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 reviewed 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 was to ensure that the control room can be maintained as the backup center from which technical support center personnel can safely operate in the case of an accident. The NRC staff's review focused on the effects of the proposed EPU on 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, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with postulated accidents, including the effects of the release of toxic gases; and (2) GDC-19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent, to any part of the body, for the duration of the accident. 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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. 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 focused on the effects of the proposed EPU on system functional design, environmental design, and provisions to preclude temperatures in the adsorber section from exceeding design limits. The NRC's acceptance criteria for ESF atmosphere cleanup systems are based on (1) GDC-19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent, to any part of the body, for the duration of the accident; (2) GDC-41, insofar as it requires that systems to control fission products released into the reactor containment be provided to reduce the concentration and quality of fission products released to the environment following postulated accidents; (3) GDC-61, insofar as it requires that systems that may contain radioactivity be designed to assure adequate safety under normal and postulated accident conditions; and (4) GDC-64, insofar as it requires that means be provided for monitoring effluent discharge paths and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences (AOOs), and postulated accidents. Specific review criteria are contained in SRP Section 6.5.1.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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, AOOs, and DBA conditions. The NRC's review of the CRAVS focused on the effects that the proposed EPU will have on the functional performance of safety-related portions of the system. The review included 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, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents; (2) GDC-19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident; and (3) GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents. Specific review criteria are contained in SRP Section 9.4.1.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 a DBA under the conditions of the proposed EPU, and associated changes to parameters affecting environmental conditions for control room personnel and equipment. Accordingly, 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, AOOs, and following postulated fuel handling accidents. The NRC staff's review focused 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, insofar as it requires that the plant design include means to control the release of radioactive effluents, and (2) GDC-61, insofar as it requires that systems which contain radioactivity be designed with appropriate confinement and containment. Specific review criteria are contained in SRP Section 9.4.2.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 AOOs, and after postulated accidents. The NRC staff's review focused 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, insofar as it requires that the plant design include means to control the release of radioactive effluents. Specific review criteria are contained in SRP Sections 9.4.3 and 9.4.4.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 focused 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 covered (1) the ability of the ESF 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 (e.g., 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, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents; (2) GDC-17, insofar as it requires onsite and offsite electric power systems be provided to permit functioning of SSCs important to safety; and (3) GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents. Specific review criteria are contained in SRP Section 9.4.5.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 ESFVS 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 reviewed the fuel system to ensure that (1) the fuel system is not damaged as a result of normal operation and AOOs, (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 covered fuel system damage mechanisms, limiting values for important parameters, and performance of the fuel system during normal operation, AOOs, and postulated accidents. The NRC's acceptance criteria are based on (1) 10 CFR 50.46, insofar as it establishes standards for the calculation of emergency core cooling system (ECCS) performance and acceptance criteria for that calculated performance; (2) GDC-10, insofar as it requires that the reactor core be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs; (3) GDC-27, insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to assure the capability to cool the core is maintained; and (4) GDC-35, insofar as it requires that a system to provide abundant emergency core cooling be provided to transfer heat from the reactor core following any LOCA. 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 AOOs, (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 reviewed 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 and 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 covered 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, insofar as it requires that the reactor core be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs; (2) GDC-11, insofar as it requires that the reactor core be designed so that the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity; (3) GDC-12, insofar as it requires that the reactor core be designed to assure that power oscillations, which can result in conditions exceeding SAFDLs, are not possible or can be reliably and readily detected and suppressed; (4) GDC-13, insofar as it requires that instrumentation and controls be provided to monitor variables and systems affecting the fission process over anticipated ranges for normal operation, AOOs and accident conditions, and to maintain the variables and systems within prescribed operating ranges; (5) GDC-20, insofar as it requires that the protection system be designed to initiate the reactivity control systems automatically to assure that acceptable fuel design limits are not exceeded as a result of AOOs and to automatically initiate operation of systems and components important to safety under accident conditions; (6) GDC-25, insofar as it requires that the protection system be designed to assure that SAFDLs are not exceeded for any single malfunction of the reactivity control systems; (7) GDC-26, insofar as it requires that two independent reactivity control systems be provided, with both systems capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes; (8) GDC-27, insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to assure the capability to cool the core is maintained; and (9) GDC-28, insofar as it requires that the reactivity control systems be designed to assure that the effects of postulated reactivity accidents can neither result in damage to the RCPB greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to significantly impair 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 reviewed 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 AOOs, and (4) is not susceptible to thermal-hydraulic instability. The review also covered hydraulic loads on the core and RCS components during normal operation and DBA 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, insofar as it requires that the reactor core be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs; and (2) GDC-12, insofar as it requires that the reactor core and associated coolant, control, and protection systems be designed to assure that power oscillations, which can result in conditions exceeding SAFDLs, 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 (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 that would lead to fuel damage during normal reactor operation and AOOs, and (4) 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 covered 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 AOOs, and prevent or mitigate the consequences of postulated accidents. The review also covered the CRDS cooling system to ensure that it will continue to meet its design requirements. The NRC's acceptance criteria are based on (1) GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents; (2) GDC-23, insofar as it requires that the protection system be designed to fail into a safe state; (3) GDC-25, insofar as it requires that the protection system be designed to assure that SAFDLs are not exceeded for any single malfunction of the reactivity control systems; (4) GDC-26, insofar as it requires that two independent reactivity control systems be provided, with both systems capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes; (5) GDC-27, insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to assure the capability to cool the core is maintained; (6) GDC-28, insofar as it requires that the reactivity control systems be designed to assure that the effects of postulated reactivity accidents can neither result in damage to the RCPB greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to significantly impair the capability to cool the core; (7) GDC-29, insofar as it requires that the protection and reactivity control systems be designed to assure an extremely high probability of accomplishing their safety functions in event of AOOs; and (8) 10 CFR 50.62(c)(3), insofar as it requires that all BWRs have an alternate rod injection (ARI) system diverse from the reactor trip system, and that the ARI system have redundant scram air header exhaust valves. Specific review criteria are contained in SRP Section 4.6.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 bases will continue to

be followed upon 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 covered 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, insofar as it requires that the RCS and associated auxiliary, control, and protection systems be designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOOs; and (2) GDC-31, insofar as it requires that the RCPB be 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 (1) adequately accounted for the effects of the proposed EPU on pressurization events and overpressure protection features and (2) 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 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 covered the effect of the proposed EPU on the functional capability of the system. The NRC's acceptance criteria are based on (1) GDC-4, insofar as it requires that SSCs important to safety be protected against dynamic effects; (2) GDC-5, insofar as it requires that SSCs important to safety not be 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, insofar as it requires that the protection and reactivity control systems be designed to assure an extremely high probability of accomplishing their safety functions in event of AOOs; (4) GDC-33, insofar as it requires that a system to provide reactor coolant makeup for protection against small breaks in the RCPB be provided so the fuel design limits are not exceeded; (5) GDC-34, insofar as it requires that a residual heat removal system be provided to transfer fission product decay heat and other residual heat from the reactor core at a rate such that SAFDLs and the design conditions of the RCPB are not exceeded; (6) GDC-54, insofar as it requires that piping systems penetrating containment be designed with the capability to periodically test the operability of the isolation valves to determine if valve leakage is within acceptable limits; and (7) 10 CFR 50.63, insofar as it requires that the plant 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 covered 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, insofar as it requires that SSCs important to safety be protected against dynamic effects; (2) GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions; and (3) 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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, 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 covered 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, insofar as it requires that two independent reactivity control systems of different design principles be provided, and that one of the systems be capable of holding the reactor subcritical in the cold condition; (2) GDC-27, insofar as it requires 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), insofar as it requires that the SLCS be capable of reliably injecting a borated water solution into the reactor pressure vessel at a boron concentration, boron enrichment, and flow rate that provides a set level of reactivity control, and **[DEPENDING ON CONSTRUCTION PERMIT DATE OR ORIGINAL DESIGN]** that the system initiate automatically. 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 covered (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) operator actions, and (7) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC-10, insofar as it requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations including AOOs; (2) GDC-15, insofar as it requires that the RCS and its associated auxiliary systems be designed with margin sufficient to ensure that the design condition of the RCPB are not exceeded during any condition of normal operation; (3) GDC-20, insofar as it requires that the reactor protection system be designed to initiate automatically the operation of appropriate systems, including the reactivity control systems, to ensure that SAFDLs are not exceeded during any condition of normal operation, including AOOs; and (4) GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operation, including AOOs, SAFDLs are not exceeded. 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 SAFDLs 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 covered 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, insofar as it requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including AOOs; (2) GDC-15, insofar as it requires that the RCS and its associated auxiliary systems be designed with margin sufficient to ensure that the design condition of the RCPB are not exceeded during any condition of normal operation; and (3) GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operation, including AOOs, SAFDLs are not exceeded. 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 SAFDLs 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 covered (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, insofar as it requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including AOOs; (2) GDC-15, insofar as it requires that the RCS and its associated auxiliary systems be designed with margin sufficient to ensure that the design condition of the RCPB are not exceeded during any condition of normal operation; and (3) GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operation, including AOOs, SAFDLs are not exceeded. 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 SAFDLs 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 covered (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, insofar as it requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including AOOs; (2) GDC-15, insofar as it requires that the RCS and its associated auxiliary systems be designed with margin sufficient to ensure that the design condition of the RCPB are not exceeded during any condition of normal operation; and (3) GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operation, including AOOs, SAFDLs are not exceeded. 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 SAFDLs 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 SAFDLs are exceeded during the transient. Reactor protection and safety systems are actuated to mitigate the transient. The NRC staff's review covered (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) operator actions, and (7) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC-10, insofar as it requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including AOOs; (2) GDC-15, insofar as it requires that the RCS and its associated auxiliary systems be designed with margin sufficient to ensure that the design condition of the RCPB are not exceeded during any condition of normal operation; and (3) GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operation, including AOOs, SAFDLs are not exceeded. 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 SAFDLs 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 covered (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) operator actions, and (7) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC-27, insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to assure the capability to cool the core is maintained; (2) GDC-28, insofar as it requires that the reactivity control systems be designed to assure that the effects of postulated reactivity accidents can neither result in damage to the RCPB greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to significantly impair the capability to cool the core; and (3) GDC-31, insofar as it requires that the RCPB be designed with margin sufficient to assure that, under specified conditions, it will behave in a nonbrittle manner and the probability of a rapidly 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 conditions 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 covered (1) the description of the causes of the transient and the transient itself, (2) the initial conditions, (3) the values of 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, insofar as it requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including AOOs; (2) GDC-20, insofar as it requires that the reactor protection system be designed to initiate automatically the operation of appropriate systems, including the reactivity control systems, to ensure that SAFDLs are not exceeded as a result of AOOs; and (3) GDC-25, insofar as it requires that the protection system be designed to assure that SAFDLs are not exceeded for any 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 necessary 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 SAFDLs 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 covered (1) the description of the causes of the AOO and the description of the event itself, (2) the initial conditions, (3) the values of 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, insofar as it requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including AOOs; (2) GDC-20, insofar as it requires that the reactor protection system be designed to initiate automatically the operation of appropriate systems, including the reactivity control systems, to ensure that SAFDLs are not exceeded as a result of AOOs; and (3) GDC-25, insofar as it requires that the protection system be designed to assure that SAFDLs are not exceeded for any 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 SAFDLs 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

Regulatory Evaluation

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 covered (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, insofar as it requires that the RCS be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs; (2) GDC-20, insofar as it requires that the protection system be designed to initiate automatically the operation of appropriate systems to ensure that SAFDLs are not exceeded as a result of operational occurrences; (3) GDC-15, insofar as it requires that the RCS and its associated auxiliary systems be designed with margin sufficient to ensure that the design condition of the RCPB are not exceeded during AOOs; (4) GDC-28, insofar as it requires that the reactivity control systems be designed to assure that the effects of postulated reactivity accidents can neither result in damage to the RCPB greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to significantly impair the capability to cool the core; and (5) GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operation, including AOOs, SAFDLs are not exceeded. 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

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 SAFDLs 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 evaluated the consequences of a control rod drop accident in the area of reactor physics. The NRC staff's review covered 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, insofar as it requires that the reactivity control systems be designed to assure that the effects of postulated reactivity accidents can neither result in damage to the RCPB greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to significantly impair the capability 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 and depressurization may result. Reactor protection and safety systems are actuated to mitigate these events. The NRC staff's review covered (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, insofar as it requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including AOOs; (2) GDC-15, insofar as it requires that the RCS and its associated auxiliary systems be designed with margin sufficient to ensure that the design conditions of the RCPB are not exceeded during AOOs; and (3) GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operation, including AOOs, SAFDLs are not exceeded. 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 SAFDLs 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 covered (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, insofar as it requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including AOOs; (2) GDC-15, insofar as it requires that the RCS and its associated auxiliary systems be designed with margin sufficient to ensure that the design conditions of the RCPB are not exceeded during AOOs; and (3) GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operation, including AOOs, SAFDLs are not exceeded. 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 SAFDLs 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 in 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 to replenish it. 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 these accidents. The NRC staff's review covered (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, and calculations of the reactor power, pressure, flow, and temperature transients; (5) calculations of 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 protection and ECCS systems; and (7) operator actions. The NRC's acceptance criteria are based on (1) 10 CFR § 50.46, insofar as it establishes standards for the calculation of ECCS performance and acceptance criteria for that calculated performance; (2) 10 CFR Part 50, Appendix K, insofar as it establishes required and acceptable features of evaluation models for heat removal by the ECCS after the blowdown phase of a LOCA; (3) GDC-4, insofar as it requires that SSCs important to safety be protected against dynamic effects associated with flow instabilities and loads such as those resulting from water hammer; (4) GDC-27, insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to assure the capability to cool the core is maintained; and (5) GDC-35, insofar as it requires that a system to provide abundant emergency core cooling be provided to transfer heat from the reactor core following any LOCA at a rate so that fuel clad damage that could interfere with continued effective core cooling will be prevented. 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 AOO 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 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 into the reactor vessel a borated water solution with reactivity control at least 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 was 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 by the proposed EPU, 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), insofar as they apply to the plant design. In addition, the NRC staff reviewed the licensee's ATWS analysis to ensure that (1) the peak vessel bottom pressure is less than the 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 evaluated 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 reviewed 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. *Insert the following sentence if the licensee relied upon generic vendor analyses* **[The NRC staff reviewed the licensee's justification of the applicability of generic vendor analyses 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 needs. The NRC staff's review covered the ability of the storage facilities to maintain the new fuel in a subcritical array during all credible storage conditions. The review focused 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, insofar as it requires the prevention of criticality in fuel storage systems by physical systems or processes, preferably utilizing geometrically safe configurations. Specific review criteria are contained in SRP Section 9.1.1.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 covered 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, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, and (2) GDC-62, insofar as it requires that criticality in the fuel storage systems be prevented by physical systems or processes, preferably by use of geometrically safe configurations. Specific review criteria are contained in SRP Section 9.1.2.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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

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2.9 Source Terms and Radiological Consequences Analyses

2.9.1 Source Terms for Radwaste Systems Analyses

Regulatory Evaluation

The NRC staff reviewed the radioactive source term associated with EPU to ensure the adequacy of the sources of radioactivity used by the licensee as input to calculations 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 included 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 radionuclides other than fission products 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, insofar as it establishes requirements for radioactivity in liquid and gaseous effluents released to unrestricted areas; (2) 10 CFR Part 50, Appendix I, insofar as it establishes numerical guides for design objectives and limiting conditions for operation to meet the "as low as is reasonably achievable" criterion; and (3) GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents. Specific review criteria are contained in SRP Section 11.1.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 reviewed 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 included (1) the sequence of events; and (2) models, assumptions, and values of parameter inputs used by the licensee for the calculation of the total effective dose equivalent (TEDE). The NRC's acceptance criteria for radiological consequences analyses using an alternative source term are based on (1) 10 CFR 50.67, insofar as it sets standards for radiological consequences of a postulated accident, and (2) GDC-19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem TEDE, as defined in 10 CFR 50.2, for the duration of the 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 reviewed the implementation of alternative source terms. The NRC's acceptance criteria for implementation of alternative source terms are based on (1) 10 CFR 50.67, insofar as it sets standards for the implementation of an alternative source term in current operating nuclear power plants; (2) 10 CFR 50.49, insofar as it requires qualification of safety-related equipment, as defined in that section, including and based on integrated radiation dose during normal and accident conditions; (3) GDC-19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem TEDE, as defined in 10 CFR 50.2, for the duration of the accident; (4) Paragraph IV.E.8 of 10 CFR Part 50, Appendix E, insofar as it requires a licensee onsite technical support center and a licensee near-site emergency operations facility from which effective direction can be given and effective control can be exercised during an emergency; and (5) 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 ESFs remain acceptable with respect to the radiological consequences of postulated DBAs since, as set forth above, 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 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 - 2.9.8 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 reviewed the analyses of the radiological consequences of a control rod drop accident (CRDA). The NRC staff's review included 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, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident, and (2) 10 CFR Part 100, insofar as it establishes requirements for assuring that radiological doses from postulated accidents will be acceptably low. 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 reviewed 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 included (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, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent, to any part of the body, for the duration of the accident, and (2) GDC-55, insofar as it establishes isolation requirements for small-diameter lines connected to the primary system that form 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 will 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 reviewed the analyses of the radiological consequences of an MSLB accident outside the containment to ensure that radioactive releases due to such an event are adequately limited by the TS limit on primary coolant activity. The NRC staff's review included two cases for the reactor coolant iodine concentration: (1) an MSLB with a preaccident iodine spike and (2) an MSLB 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, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident, and (2) 10 CFR Part 100, insofar as it establishes requirements for assuring that radiological doses from postulated accidents will be acceptably low. 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 reviewed the analyses of the radiological consequences of a design-basis LOCA. This review included 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 included (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 were 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, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident, and (2) 10 CFR Part 100, insofar as it establishes requirements for assuring that radiological doses from postulated accidents will be acceptably low. 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 reviewed the analyses of the radiological consequences of a postulated FHA. The purpose of this review was 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 included (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, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent, to any part of the body, for the duration of the accident; (2) GDC-61, insofar as it requires that systems that contain radioactivity be designed with appropriate containment, confinement, and filtering systems; and (3) 10 CFR Part 100, insofar as it establishes requirements for assuring that radiological doses from postulated accidents will be acceptably low. 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 reviewed 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 was conducted to verify various design and operational aspects of the system. The NRC staff's review included (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 atmosphere 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, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident; (2) GDC-61, insofar as it requires that systems that contain radioactivity be designed with appropriate containment, confinement, and filtering systems; and (3) 10 CFR Part 100, insofar as it establishes requirements for assuring that radiological doses from postulated accidents will be acceptably low. 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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]

INSERT 10

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 conducted 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 included 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 evaluated how personnel doses needed to access plant vital areas following an accident are affected. The NRC staff considered 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 considered 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 was conducted to ensure that operator performance is not adversely affected as a result of system changes made to implement the proposed EPU. The NRC staff's review covered changes to operator actions, human-system interfaces, and procedures and training needed for the proposed EPU. The NRC's acceptance criteria for human factors are based on GDC-19, 10 CFR 50.120, 10 CFR Part 55, and the guidance in 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 evaluation of the responses.

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 NRC staff statement on why the response is acceptable]

2. Changes to Operator Actions Sensitive to Power Uprate

Describe any new operator actions needed 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 involve 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 NRC staff statement on why the response is acceptable]

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 will be tested to determine they could use the instruments reliably. (SRP Section 18.0)

[Insert licensee's response followed by NRC staff statement on why the response is acceptable]

4. Changes on the Safety Parameter Display System

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

[Insert licensee's response followed by NRC staff statement on why the response is acceptable]

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

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

[Insert licensee's response followed by NRC staff statement on why the response is acceptable]

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 50.120, 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 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 NRC's acceptance criteria for the proposed EPU test program are based on 10 CFR Part 50, Appendix B, Criterion XI, which requires 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 proposed EPU, or modified to support the proposed EPU, 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 EPU

Regulatory Evaluation

The licensee conducted a risk evaluation 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 present if any issue would potentially rebut the presumption of adequate protection provided by the licensee to meet the deterministic requirements and regulations. The NRC staff’s review covered 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 covered the quality of the risk analyses used by the licensee to support the application for the proposed EPU. This included a review of the licensee’s 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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**

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Attachment: List of Acronyms

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 design basis of the plant. Where differences exist between the plant-specific design basis and RS-001, the licensee described the differences and provided evaluations consistent with the design 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 necessary to implement 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 design-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 the affects of the 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

As described above, 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's 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 EPU's, the extent and unique nature of modifications necessary to implement the proposed EPU, and new conditions of operation necessary 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. *If no comments were received, use the following sentence: [No comments were received on the draft Environmental Assessment.] If comments were received, use the following sentence: [The NRC staff received comments which were addressed in the final environmental assessment.]* 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, Revision 0, "Review Standard for Extended Power Uprates," December 2003.
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	<i>Code of Federal Regulations</i>
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 (or criteria)
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 focused 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, insofar as it requires that the reactor coolant pressure boundary (RCPB) be designed, fabricated, erected, and tested so as to have an extremely low probability of rapidly propagating; (2) GDC-31, insofar as it requires that the RCPB be designed with margin sufficient to assure that, under specified conditions, it will behave in a nonbrittle manner and the probability of a rapidly propagating fracture is minimized; (3) 10 CFR Part 50, Appendix H, which provides for monitoring changes in the fracture toughness properties of materials in the reactor vessel beltline region; and (4) 10 CFR 50.60, which requires 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 10 CFR Part 50, Appendix H, and 10 CFR 50.60, and will provide the licensee with information to ensure continued compliance with GDC-14 and GDC-31 in this respect 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 covered the P-T limits methodology and the calculations for the number of effective full power years specified for the proposed EPU, 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, insofar as it requires that the RCPB be designed, fabricated, erected, and tested so as to have an extremely low probability of rapidly propagating fracture; (2) GDC-31, insofar as it requires that the RCPB be designed with margin sufficient to assure that, under specified conditions, it will behave in a nonbrittle manner and the probability of a rapidly propagating fracture is minimized; (3) 10 CFR Part 50, Appendix G, which specifies fracture toughness requirements for ferritic components of the RCPB; and (4) 10 CFR 50.60, which requires 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 operation under the proposed EPU conditions. Based on this, the NRC staff concludes that the proposed P-T limits will continue to meet the requirements of 10 CFR Part 50, Appendix G, and 10 CFR 50.60 and will enable the licensee to comply with GDC-14 and GDC-31 in this respect 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 covered 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, insofar as it requires that the RCPB be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating fracture, and of gross rupture; (2) GDC-31, insofar as it requires that the RCPB be designed with margin sufficient to assure that, under specified conditions, it will behave in a nonbrittle manner and the probability of a rapidly propagating fracture is minimized; and (3) 10 CFR 50.61, insofar as it sets fracture toughness criteria for protection against 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 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 covered 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 covered 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) 10 CFR 50.55a and GDC-1, insofar as they require that SSCs important to safety be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety functions to be performed; (2) GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents; (3) GDC-14, insofar as it requires that the RCPB be designed, fabricated, erected, and tested so as to have an extremely low probability of rapidly propagating fracture; (4) GDC-31, insofar as it requires that the RCPB be designed with margin sufficient to assure that, under specified conditions, it will behave in a nonbrittle manner and the probability of a rapidly propagating fracture is minimized; and (5) 10 CFR Part 50, Appendix G, which specifies fracture toughness requirements for ferritic components 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 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 covered (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) deterministic fracture mechanics and leak detection methods. The NRC's acceptance criteria for LBB are based on GDC-4, insofar as it allows for exclusion of dynamic effects of postulated pipe ruptures from the design basis. 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 covered protective coating systems used inside the containment for their suitability for and stability under design-basis loss-of-coolant accident (DBLOCA) 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, which states quality assurance requirements for the design, fabrication, and construction of safety-related SSCs and (2) Regulatory Guide 1.54, Revision 1, for guidance on 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 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.

2.1.8 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. 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 due to FAC depend on velocity of flow, fluid 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 has reviewed 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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. The NRC staff further 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 covered the effects of changes in differential pressure, temperature, and flow rates resulting from the proposed EPU on plugging limits, potential degradation mechanisms (e.g., flow-induced vibration), 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 requirements 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]**, all of which form the basis for alternate repair criteria or redefined inspection boundaries.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 continued acceptability of the plant's TSs under the proposed EPU conditions and has identified appropriate degradation management inspections to address the effects of changes in temperature, differential pressure, and flow rates on SG tube integrity. The NRC staff further concludes that the licensee has demonstrated that SG tube integrity will continue to be maintained and will continue to meet the performance criteria in NEI 97-06 and the requirements of 10 CFR 50.55a 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 SG tubes. The SG blowdown system (SGBS) provides a means for removing SG secondary-side impurities and thus, assists in maintaining acceptable secondary-side water chemistry in the SGs. The design basis of the SGBS includes consideration of expected and design flows for all modes of operation. The NRC staff's review covered the ability of the SGBS to remove particulate and dissolved impurities from the SG 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, insofar as it requires that the RCPB be designed so as to have an extremely low probability of abnormal leakage, of rapidly propagating fracture, and of gross rupture. Specific review criteria are contained in SRP Section 10.4.8.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 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, insofar as it requires that the RCPB be designed so as to have an extremely low probability of abnormal leakage, of rapidly propagating fracture, and of gross rupture, and (2) GDC-29, insofar as it requires that the reactivity control systems be designed to assure an extremely high probability of accomplishing their safety functions in event of anticipate operational occurrences. Specific review criteria are contained in SRP Section 9.3.4.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 Additional Review Areas (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 conducted 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 covered (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 ISI programs or the use of special protective devices such as pipe-whip restraints, (3) 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 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, which requires SSCs important to safety to be 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 has reviewed 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 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 covered (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 included 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, insofar as they require that SSCs important to safety be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety functions to be performed; (2) GDC 2, insofar as it requires that SSCs important to safety be designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions; (3) GDC 4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents; (4) GDC-14, insofar as it requires that the RCPB be designed, fabricated, erected, and tested so as to have an extremely low probability of rapidly propagating fracture; and (5) GDC 15, insofar as it requires that the RCS be designed with margin sufficient to ensure that the design conditions of the RCPB are not exceeded during any condition of normal operation. 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 (NSSS) piping, components, and supports. Include an intermediate conclusion in the form of "Because [summarize reasons], the NSSS piping, components, and supports are adequate under the proposed EPU conditions."]

Balance-of-Plant Piping, Components, and Supports

[Insert technical evaluation for balance-of-plant piping, components, and supports. Include an intermediate conclusion in the form of "Because [summarize reasons], the balance-of-plant piping, components, and supports are adequate under the proposed EPU conditions."]

Reactor Vessel and Supports

[Insert technical evaluation for reactor vessel and supports. Include an intermediate conclusion in the form of "Because [summarize reasons], the reactor vessel and supports are adequate under the proposed EPU conditions."]

Control Rod Drive Mechanism

[Insert technical evaluation for control rod drive mechanism. Include an intermediate conclusion in the form of “Because [summarize reasons], the control rod drive mechanism is adequate under the proposed EPU conditions.”]

Steam Generators and Supports

[Insert technical evaluation for SGs and supports. Include an intermediate conclusion in the form of “Because [summarize reasons], the SG and supports are adequate under the proposed EPU conditions.”]

Reactor Coolant Pumps and Supports

[Insert technical evaluation for reactor coolant pumps and supports. Include an intermediate conclusion in the form of “Because [summarize reasons], the reactor coolant pumps and supports are adequate under the proposed EPU conditions.”]

Pressurizer and Supports

[Insert technical evaluation for pressurizer and supports. Include an intermediate conclusion in the form of “Because [summarize reasons], the pressurizer and supports are adequate under the proposed EPU conditions.”]

Conclusion

The NRC staff has reviewed the licensee’s evaluations related to the structural integrity of pressure-retaining components and their supports. For the reasons set forth above, the NRC staff concludes that the licensee has adequately addressed the effects of the proposed EPU on these components and their supports. Based on the above, 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 reviewed 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 loss-of-coolant accidents (LOCAs), and the identification of design transient occurrences. The NRC staff's review covered (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 included a comparison of the resulting stresses and CUFs against the corresponding Code-allowable limits. The NRC's acceptance criteria are based on (1) 10 CFR 50.55a and GDC-1, insofar as they require that SSCs important to safety be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety functions to be performed; (2) GDC-2, insofar as it requires that SSCs important to safety be designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions; (3) GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents; and (4) GDC-10, insofar as it requires that the reactor core be designed 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 the reactor internals and core supports. 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 included 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 focused on the effects of the proposed EPU on the required functional performance of the valves and pumps. The review also covered 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 evaluated 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, insofar as it requires that SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed; (2) GDC 37, GDC 40, GDC 43, and GDC 46, insofar as they require that the ECCS, the containment heat removal system, the containment atmospheric cleanup systems, and the cooling water system, respectively, be designed to permit appropriate periodic testing to ensure the leak-tight integrity and performance of their active components; (3) GDC-54, insofar as it requires that piping systems penetrating containment be designed with the capability to periodically test the operability of the isolation valves to determine if valve leakage is within acceptable limits; and (4) 10 CFR 50.55a(f), insofar as it requires that pumps and valves subject to that section must meet the inservice testing program requirements identified in that section. 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the licensee's assessments 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 safety-related pumps and valves. 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 to preventing significant releases of radioactive materials to the environment are also covered by this section. The NRC staff's review focused 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, insofar as it requires that SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed; (2) GDC-30, insofar as it requires that components that are part of the RCPB be designed, fabricated, erected, and tested to the highest quality standards practical; (3) GDC-2, insofar as it requires that SSCs important to safety be designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions; (4) 10 CFR Part 100, Appendix A, which sets forth the principal seismic and geologic considerations for the evaluation of the suitability of plant design bases established in consideration of the seismic and geologic characteristics of the plant site; (5) GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents; (6) GDC-14, insofar as it requires that the RCPB be designed, fabricated, erected, and tested so as to have an extremely low probability of rapidly propagating fracture; and (7) 10 CFR Part 50, Appendix B, which sets quality assurance requirements for safety-related equipment. Specific review criteria are contained in SRP Section 3.10.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 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 was 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, which sets forth requirements for 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 electrical equipment. The NRC staff further concludes that the electrical equipment will continue to meet 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 covered 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 focused on whether the loss of the nuclear unit, the largest operating unit on the grid, or the most critical transmission line will result in the loss of offsite power (LOOP) 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 alternating current (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 covered 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, insofar as it requires the system to have the capacity and capability to perform its intended functions during anticipated operational occurrences and accident conditions. Specific review criteria are contained in SRP Sections 8.1 and 8.3.1.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 direct current (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 covered 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, insofar as it requires the system to have the capacity and capability to perform its intended functions during anticipated operational occurrences and accident conditions. Specific review criteria are contained in SRP Sections 8.1 and 8.3.2

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 LOOP 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 focused on the impact of the proposed EPU on the plant's ability to cope with and recover from an 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 an 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 (including 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 conducted 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 necessary for the proposed EPU are adequately designed such that the systems continue to meet their safety functions. The NRC staff's review was 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 necessary to achieve the proposed EPU are consistent with the plant's design 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 conducted a review in the area of flood protection to ensure that SSCs important to safety are protected from flooding. The NRC staff's review covered flooding of SSCs important to safety from internal sources, such as those caused by failures of tanks and vessels. The NRC staff's review focused 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 an uncontaminated drainage system. The NRC staff's review of the EFDS included the collection and disposal of liquid effluents outside containment.

The NRC staff's review focused on any changes in fluid volumes or pump capacities that are necessary 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 insofar as they require the EFDS to be designed 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 focused on changes in flooding analyses that are necessary due to increases in fluid volumes or installation of larger capacity pumps or piping needed 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 covered pressurized components and systems, and high-speed rotating machinery. The NRC staff's review was conducted to ensure that safety-related SSCs are adequately protected from internally generated missiles. In addition, for cases where safety-related SSCs are located in areas containing non-safety-related SSCs, the NRC staff reviewed 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 focused 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 focused 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, and relates to 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 conducted 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 included high and moderate energy fluid system piping located outside of containment. The NRC staff's review focused on the effects of pipe failures on plant 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, which requires, in part, that SSCs important to safety be 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

Conclusion

The NRC staff has reviewed the changes that are necessary 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 the dynamic effects of 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 focused 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 will 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, insofar as they require the development of an FPP to ensure, among other things, the capability to safely shut down the plant; (2) GDC-3, insofar as it requires that (a) SSCs important to safety be designed and located to minimize the probability and effect of fires, (b) noncombustible and heat resistant materials be used, and (c) fire detection and fighting systems be provided and designed to minimize the adverse effects of fires on SSCs important to safety; (3) GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions. Specific review criteria are contained in SRP Section 9.5.1, as supplemented by the guidance provided in Attachment 2 to Matrix 5 of Section 2.1 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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, Appendix R to 10 CFR Part 50, 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 tank is design with a capacity 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 conducted 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 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 and the tank rupture disk relief capacity should be adequate, taking into consideration the capacity of the pressurizer power-operated relief and safety valves; the piping to the tank should be adequately sized; and systems inside containment should 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: (1) GDC-2, insofar as it requires that SSCs important to safety be designed to withstand the effects of earthquakes; and (2) GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate and be compatible with specified environmental conditions, and be appropriately protected against dynamic effects, including the effects of missiles. Specific review criteria are contained in SRP Section 5.4.11.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 covered 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 focused 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, insofar as it requires that the containment atmosphere cleanup system be provided to reduce the concentration of fission products released to the environment following postulated accidents. Specific review criteria are contained in SRP Section 6.5.3.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 in fission products and changes in expected environmental conditions that would result from the proposed EPU. 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 focused 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, insofar as it requires that the plant design include means to control the release of radioactive effluents; and (2) GDC-64, insofar as it requires that means be provided for monitoring effluent discharge paths and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences and postulated accidents. Specific review criteria are contained in SRP Section 10.4.2.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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.

2.5.3.3 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 reviewed 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, insofar as it requires that the plant design include means to control the release of radioactive effluents; and (2) GDC-64, insofar as it requires that means be provided for monitoring effluent discharge paths and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences and postulated accidents. Specific review criteria are contained in SRP Section 10.4.3.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 the proposed EPU focused 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, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions, (2) GDC-44, insofar as it requires that a system with the capability to transfer heat loads from safety-related SSCs to a heat sink under both normal operating and accident conditions be provided, and (3) GDC-61, insofar as it requires that fuel storage systems be designed with RHR capability reflecting the importance to safety of decay heat removal, 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 1 to Matrix 5 of Section 2.1 of RS-001.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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.

2.5.4.2 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 covered the characteristics of the station SWS components with respect to their functional performance as affected by adverse operational (i.e., water hammer) conditions, abnormal operational conditions, and accident conditions (e.g., a LOCA with the LOOP). The NRC staff's review focused on the additional heat load that would result from the proposed EPU. The NRC's acceptance criteria are based on (1) GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, including flow instabilities and loads (e.g., water hammer), maintenance, testing, and postulated accidents; (2) GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions; and (3) GDC-44, insofar as it requires that a system with the capability to transfer heat loads from safety-related SSCs to a heat sink under both normal operating and accident conditions be provided. 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 on system performance that would result from the proposed EPU. 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 covered reactor auxiliary cooling water systems that are required for (1) safe shutdown during normal operations, anticipated operational occurrences, and 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 covered 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 was placed on the cooling water systems for safety-related components (e.g., ECCS equipment, ventilation equipment, and reactor shutdown equipment). The NRC staff's review focused on the additional heat load that would result from the proposed EPU. The NRC's acceptance criteria for the reactor auxiliary cooling water system are based on (1) GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation including flow instabilities and attendant loads (i.e., water hammer), maintenance, testing, and postulated accidents; (2) GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions; and (3) GDC-44, insofar as it requires that a system with the capability to transfer heat loads from safety-related SSCs to a heat sink under both normal operating and accident conditions be provided. 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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.

2.5.4.4 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 focused on the impact that the proposed EPU has on the decay heat removal capability of the UHS. Additionally, the NRC staff's review included 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, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety; and (2) GDC-44, insofar as it requires that a system with the capability to transfer heat loads from safety-related SSCs to a heat sink under both normal operating and accident conditions be provided. Specific review criteria are contained in SRP Section 9.2.5.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 following implementation of the proposed EPU. 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 an SBO. The NRC staff's review for the proposed EPU focused 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 considered the effects of the proposed EPU on the likelihood of creating 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 AFWS are based on (1) GDC-4, insofar as it requires that SSCs important to safety be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids that may result from equipment failures; (2) GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions; (3) GDC-19, insofar as it requires that equipment at appropriate locations outside the control room be provided with (a) the capability for prompt hot shutdown of the reactor, and (b) a potential capability for subsequent cold shutdown of the reactor; (4) GDC-34, insofar as it requires that an RHR system be provided to transfer fission product decay heat and other residual heat from the reactor core, and that suitable isolation be provided to assure that the system safety function can be accomplished, assuming a single failure; and (5) GDC-44, insofar as it requires that a system with the capability to transfer heat loads from safety-related SSCs to a heat sink under both normal operating and accident conditions be provided, and that suitable isolation be provided to assure that the system safety function can be accomplished, assuming a single failure. Specific review criteria are contained in SRP Section 10.4.9.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 SGs 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 NSSS to the power conversion system and various safety-related and non-safety-related auxiliaries. The NRC staff's review focused 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, insofar as it requires that SSCs important to safety be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids that may result from equipment failures; (2) GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions; and (3) GDC-34, insofar as it requires that an RHR system be provided to transfer fission product decay heat and other residual heat from the reactor core. Specific review criteria are contained in SRP Section 10.3.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 5. 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 (TBS). The NRC staff's review focused 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 TBS. The NRC's acceptance criteria for the MC system are based on GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents. Specific review criteria are contained in SRP Section 10.4.1. Specific review criteria are contained in SRP Section 10.4.1.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 TBS 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 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 SG pressure. The NRC staff's review focused 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, insofar as it requires that SSCs important to safety be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids that may result from equipment failures; and (2) GDC-34, insofar as it requires that an RHR system be provided to transfer fission product decay heat and other residual heat from the reactor core at a rate such that SAFDLs and the design conditions of the RCPB are not exceeded. Specific review criteria are contained in SRP Section 10.4.4.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 appropriate 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 SGs up to and including the outermost containment isolation valve. The NRC staff's review focused 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's safety function. The NRC staff's review also considered the effects of the proposed EPU on the feedwater system, including the AFWS piping entering the SG, 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, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, and that such SSCs be protected against dynamic effects; (2) GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions; and (3) GDC-44, insofar as it requires that a system with the capability to transfer heat loads from safety-related SSCs to a heat sink under both normal operating and accident conditions be provided, and that suitable isolation be provided to assure that the system safety function can be accomplished, assuming a single failure. Specific review criteria are contained in SRP Section 10.4.7.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 the condenser air removal system, the steam generator blowdown flash tank, and the containment purge exhausts; and the building ventilation system exhausts. The NRC staff's review focused on the effects that the proposed EPU may have on (1) the design criteria of the gaseous waste management systems, (2) methods of treatment, (3) expected releases, (4) principal parameters used in calculating the releases of radioactive materials in gaseous effluents, and (5) design features for precluding 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, insofar as it provides for demonstrating that annual average concentrations of radioactive materials released at the boundary of the unrestricted area do not exceed specified values; (2) GDC-3, insofar as it requires that (a) SSCs important to safety be designed and located to minimize the probability and effect of fires, (b) noncombustible and heat resistant materials be used, and (c) fire detection and fighting systems be provided and designed to minimize the adverse effects of fires on SSCs important to safety; (3) GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents; (4) GDC-61, insofar as it requires that systems that contain radioactivity be designed with appropriate confinement; and (5) 10 CFR Part 50 Appendix I, Sections II.B, II.C, and II.D, which set 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 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.

2.5.6.2 Liquid Waste Management Systems

Regulatory Evaluation

The NRC staff's review for liquid waste management systems focused on the effects that the proposed EPU may 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, insofar as it provides for demonstrating that annual average concentrations of radioactive materials released at the boundary of the unrestricted area do not exceed specified values; (2) GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents; (3) GDC-61, insofar as it requires that systems that contain radioactivity be designed with appropriate confinement; and (4) 10 CFR Part 50, Appendix I, Sections II.A and II.D, which set 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 for the solid waste management systems (SWMS) focused on the effects that the proposed EPU may 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, insofar as it provides for demonstrating that annual average concentrations of radioactive materials released at the boundary of the unrestricted area do not exceed specified values; (2) GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents; (3) GDC-63, insofar as it requires that systems be provided in waste handling areas to detect conditions that may result in excessive radiation levels, (4) GDC-64, insofar as it requires that means be provided for monitoring effluent discharge paths and the plant environs for radioactivity that may be released from normal operations, including AOOs, and postulated accidents; and (5) 10 CFR Part 71, which states requirements for radioactive material packaging. Specific review criteria are contained in SRP Section 11.4.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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.

2.5.7 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 supplies of sufficient capacity to perform their safety functions (e.g., power diesel engine-driven generator sets), assuming a single failure. The NRC staff's review focused on increases in emergency diesel generator electrical demand and the resulting increase in the amount of fuel oil necessary for the system to perform its safety function. The NRC's acceptance criteria for the emergency diesel engine fuel oil storage and transfer system are based on (1) GDC-4, insofar as it requires that SSCs important to safety be protected against dynamic effects, including missiles, pipe whip, and jet impingement forces associated with pipe breaks; (2) GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions; and (3) GDC-17, insofar as it requires onsite power supplies to have sufficient independence and redundancy to perform their safety functions, assuming a single failure. Specific review criteria are contained in SRP Section 9.5.4.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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.

2.5.7.2 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 and the loading of spent fuel into shipping casks. The NRC staff's review covered the avoidance of criticality accidents, radioactivity releases resulting from damage to irradiated fuel, and unacceptable personnel radiation exposures. The NRC staff's review 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, insofar as it requires that systems that contain radioactivity be designed with appropriate confinement and with suitable shielding for radiation protection; and (2) GDC-62, insofar as it requires that criticality be prevented. Specific review criteria are contained in SRP Section 9.1.4.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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.

[2.5.8 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.

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 covered 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) GDC-16, insofar as it requires that reactor containment be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment; (2) GDC-50, insofar as it requires that the containment and its internal components be 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-38, insofar as it requires that 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; (4) GDC-13, insofar as it requires that instrumentation be provided to monitor variables and systems over their anticipated ranges for normal operation and accident conditions; and (5) GDC-64, insofar as it requires that means be provided for monitoring the plant environs 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 covered 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) GDC-16, insofar as it requires that reactor containment be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment; (2) GDC-50, insofar as it requires that the containment and its internal components be 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-38, insofar as it requires that 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; (4) GDC-13, insofar as it requires that instrumentation be provided to monitor variables and systems over their anticipated ranges for normal operation and accident conditions; and (5) GDC-64, insofar as it requires that means be provided for monitoring the plant environs 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 covered the determination of the design differential pressure values for containment subcompartments. The NRC staff's review focused on the effects of the increase in mass and energy release into the containment due to operation at EPU conditions, and the resulting increase in pressurization. The NRC's acceptance criteria for subcompartment analyses are based on (1) GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, and that such SSCs be protected against dynamic effects, and (2) GDC-50, insofar as it requires that containment subcompartments be designed with sufficient margin to prevent fracture of the structure due to the calculated pressure differential conditions across the walls of the subcompartments. Specific review criteria are contained in SRP Section 6.2.1.2.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 containment SSCs important to safety will continue to be protected from the dynamic effects resulting from 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, 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 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 covered 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, insofar as it requires that sufficient conservatism be provided in the mass and energy release analysis to assure that containment design margin is maintained and (2) 10 CFR Part 50, Appendix K, insofar as it identifies sources of energy during a LOCA. Specific review criteria are contained in SRP Section 6.2.1.3.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 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.

2.6.3.2 Mass and Energy Release Analysis for Secondary System Pipe Ruptures

Regulatory Evaluation

The NRC staff's review covered 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, insofar as it requires that the margin in the design of the containment structure reflect consideration of the effects of potential energy sources that have not been included in the determination of peak conditions, the experience and experimental data available for defining accident phenomena and containment response, and the conservatism of the model and the values of input parameters. Specific review criteria are contained in SRP Section 6.2.1.4.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 (i.e., that the analysis includes sufficient margin). 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 reactions 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 covered (1) the production and accumulation of 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 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, insofar as it requires that plants be provided with the capability for controlling combustible gas concentrations in the containment atmosphere; (2) GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions; (3) GDC-41, insofar as it requires that systems be 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; (4) GDC-42, insofar as it requires that systems required by GDC-41 be designed to permit appropriate periodic inspection; and (5) GDC-43, insofar as it requires that systems required by GDC-41 be designed to permit appropriate periodic testing. ***[Include the following sentence for PWRs with ice condenser containments: Additional requirements based on 10 CFR 50.44 for control of combustible gas apply to plants with an ice condenser type of containment that do not rely on an inerted atmosphere to control hydrogen inside the containment.]*** Specific review criteria are contained in SRP Section 6.2.5.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 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 focused 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, insofar as it requires that the containment heat removal system be 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 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 reviewed 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 covered 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, insofar as it requires the use of 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 following implementation of the proposed EPU. 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 reviewed 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 was to ensure that the control room can be maintained as the backup center from which technical support center personnel can safely operate in the case of an accident. The NRC staff's review focused on the effects of the proposed EPU on 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, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with postulated accidents, including the effects of the release of toxic gases; and (2) GDC-19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent, to any part of the body, for the duration of the accident. 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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. 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., 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 focused on the effects of the proposed EPU on system functional design, environmental design, and provisions to preclude temperatures in the adsorber section from exceeding design limits. The NRC's acceptance criteria for the ESF atmosphere cleanup systems are based on (1) GDC-19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent, to any part of the body, for the duration of the accident; (2) GDC 41, insofar as it requires that systems to control fission products released into the reactor containment be provided to reduce the concentration and quality of fission products released to the environment following postulated accidents; (3) GDC-61, insofar as it requires that systems that may contain radioactivity be designed to assure adequate safety under normal and postulated accident conditions; and (4) GDC-64, insofar as it requires that means shall be provided for monitoring effluent discharge paths and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences (AOOs), and postulated accidents. Specific review criteria are contained in SRP Section 6.5.1.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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, AOOs, and DBA conditions. The NRC's review of the CRAVS focused on the effects that the proposed EPU will have on the functional performance of safety-related portions of the system. The review included 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, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents; (2) GDC-19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident; and (3) GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents. Specific review criteria are contained in SRP Section 9.4.1.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 a DBA under the conditions of the proposed EPU, and associated changes to parameters affecting environmental conditions for control room personnel and equipment. Accordingly, 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, AOOs, and following postulated fuel handling accidents. The NRC staff's review focused 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, insofar as it requires that the plant design include means to control the release of radioactive effluents, and (2) GDC-61, insofar as it requires that systems which contain radioactivity be designed with appropriate confinement and containment. Specific review criteria are contained in SRP Section 9.4.2.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 AOOs, and after postulated accidents. The NRC staff's review focused 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, insofar as it requires that the plant design include means to control the release of radioactive effluents. Specific review criteria are contained in SRP Sections 9.4.3 and 9.4.4.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 focused 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 covered (1) the ability of the ESF 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 (e.g., 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, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents; (2) GDC-17, insofar as it requires onsite and offsite electric power systems be provided to permit functioning of SSCs important to safety; and (3) GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents. Specific review criteria are contained in SRP Section 9.4.5.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 ESFVS 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 reviewed the fuel system to ensure that (1) the fuel system is not damaged as a result of normal operation and AOOs, (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 covered fuel system damage mechanisms, limiting values for important parameters, and performance of the fuel system during normal operation, AOOs, and postulated accidents. The NRC's acceptance criteria are based on (1) 10 CFR 50.46, insofar as it establishes standards for the calculation of ECCS performance and acceptance criteria for that calculated performance; (2) GDC-10, insofar as it requires that the reactor core be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs; (3) GDC-27, insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to assure the capability to cool the core is maintained; and (4) GDC-35, insofar as it requires that a system to provide abundant emergency core cooling be provided to transfer heat from the reactor core following any LOCA. 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 AOOs, (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 reviewed 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 and 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 covered 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, insofar as it requires that the reactor core be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs; (2) GDC-11, insofar as it requires that the reactor core be designed so that the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity; (3) GDC-12, insofar as it requires that the reactor core be designed to assure that power oscillations, which can result in conditions exceeding SAFDLs, are not possible or can be reliably and readily detected and suppressed; (4) GDC-13, insofar as it requires that instrumentation and controls be provided to monitor variables and systems affecting the fission process over anticipated ranges for normal operation, AOOs and accident conditions, and to maintain the variables and systems within prescribed operating ranges; (5) GDC-20, insofar as it requires that the protection system be designed to initiate the reactivity control systems automatically to assure that acceptable fuel design limits are not exceeded as a result of AOOs and to automatically initiate operation of systems and components important to safety under accident conditions; (6) GDC-25, insofar as it requires that the protection system be designed to assure that SAFDLs are not exceeded for any single malfunction of the reactivity control systems; (7) GDC-26, insofar as it requires that two independent reactivity control systems be provided, with both systems capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes; (8) GDC-27, insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to assure the capability to cool the core is maintained; and (9) GDC-28, insofar as it requires that the reactivity control systems be designed to assure that the effects of postulated reactivity accidents can neither result in damage to the RCPB greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to significantly impair 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 reviewed 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 AOOs, and (4) is not susceptible to thermal-hydraulic instability. The review also covered hydraulic loads on the core and RCS components during normal operation and DBA 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, insofar as it requires that the reactor core be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs; and (2) GDC-12, insofar as it requires that the reactor core and associated coolant, control, and protection systems be designed to assure that power oscillations, which can result in conditions exceeding SAFDLs, 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 (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 that would lead to fuel damage during normal reactor operation and AOOs, and (4) 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 covered 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 AOOs, and prevent or mitigate the consequences of postulated accidents. The review also covered the CRDS cooling system to ensure that it will continue to meet its design requirements. The NRC's acceptance criteria are based on (1) GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents; (2) GDC-23, insofar as it requires that the protection system be designed to fail into a safe state; (3) GDC-25, insofar as it requires that the protection system be designed to assure that SAFDLs are not exceeded for any single malfunction of the reactivity control systems; (4) GDC-26, insofar as it requires that two independent reactivity control systems be provided, with both systems capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes; (5) GDC-27, insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to assure the capability to cool the core is maintained; (6) GDC-28, insofar as it requires that the reactivity control systems be designed to assure that the effects of postulated reactivity accidents can neither result in damage to the RCPB greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to significantly impair the capability to cool the core; and (7) GDC-29, insofar as it requires that the protection and reactivity control systems be designed to assure an extremely high probability of accomplishing their safety functions in event of AOOs. Specific review criteria are contained in SRP Section 4.6.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 bases will continue to be followed upon 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 covered 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, insofar as it requires that the RCS and associated auxiliary, control, and protection systems be designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOOs and (2) GDC-31, insofar as it requires that the RCPB be 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 (1) adequately accounted for the effects of the proposed EPU on pressurization events and overpressure protection features and (2) 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 covered 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, insofar as it requires that the RCS and associated auxiliary, control, and protection systems be designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOOs; and (2) GDC-31, insofar as it requires that the RCPB be 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 (1) adequately accounted for the effects of the proposed EPU on pressurization events and overpressure protection features and (2) 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 covered 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, insofar as it requires that SSCs important to safety be protected against dynamic effects; (2) GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions; and (3) 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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, 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 covered (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) operator actions, and (7) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC-10, insofar as it requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations including AOOs; (2) GDC-15, insofar as it requires that the RCS and its associated auxiliary systems be designed with sufficient margin to ensure that the design condition of the RCPB are not exceeded during any condition of normal operation; (3) GDC-20, insofar as it requires that the reactor protection system be designed to initiate automatically the operation of appropriate systems, including the reactivity control systems, to ensure that SAFDLs are not exceeded during any condition of normal operation, including AOOs; and (4) GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operation, including AOOs, SAFDLs are not exceeded. 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 SAFDLs 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 covered (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) 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, insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to assure the capability to cool the core is maintained; (2) GDC-28, insofar as it requires that the reactivity control systems be designed to assure that the effects of postulated reactivity accidents can neither result in damage to the RCPB greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to significantly impair the capability to cool the core; (3) GDC-31, insofar as it requires that the RCPB be designed with sufficient margin to assure that, under specified conditions, it will behave in a nonbrittle manner and the probability of a rapidly propagating fracture is minimized; and (4) GDC-35, insofar as it requires the reactor cooling system and associated auxiliaries be 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 covered 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, insofar as it requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including AOOs; (2) GDC-15, insofar as it requires that the RCS and its associated auxiliary systems be designed with sufficient margin to ensure that the design condition of the RCPB are not exceeded during any condition of normal operation; and (3) GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operation, including AOOs, SAFDLs are not exceeded. 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 SAFDLs 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 covered (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, insofar as it requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including AOOs; (2) GDC-15, insofar as it requires that the RCS and its associated auxiliary systems be designed with sufficient margin to ensure that the design condition of the RCPB are not exceeded during any condition of normal operation; and (3) GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operation, including AOOs, SAFDLs are not exceeded. 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 SAFDLs 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 covered (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, insofar as it requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including AOOs; (2) GDC-15, insofar as it requires that the RCS and its associated auxiliary systems be designed with margin sufficient to ensure that the design condition of the RCPB are not exceeded during any condition of normal operation; and (3) GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operation, including AOOs, SAFDLs are not exceeded. 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 SAFDLs 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 covered (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) operator actions, and (7) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC-27, insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to assure the capability to cool the core is maintained; (2) GDC-28, insofar as it requires that the reactivity control systems be designed to assure that the effects of postulated reactivity accidents can neither result in damage to the RCPB greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to significantly impair the capability to cool the core; (3) GDC-31, insofar as it requires that the RCPB be designed with sufficient margin to assure that, under specified conditions, it will behave in a nonbrittle manner and the probability of a rapidly propagating fracture is minimized; and (4) GDC-35, insofar as it requires the reactor cooling system and associated auxiliaries be 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 SAFDLs are exceeded during the transient. Reactor protection and safety systems are actuated to mitigate the transient. The NRC staff's review covered (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) operator actions, and (7) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC-10, insofar as it requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including AOOs; (2) GDC-15, insofar as it requires that the RCS and its associated auxiliary systems be designed with margin sufficient to ensure that the design condition of the RCPB are not exceeded during any condition of normal operation; and (3) GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operation, including AOOs, SAFDLs are not exceeded. 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 SAFDLs 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 covered (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) operator actions, and (7) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC-27, insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to assure the capability to cool the core is maintained; (2) GDC-28, insofar as it requires that the reactivity control systems be designed to assure that the effects of postulated reactivity accidents can neither result in damage to the RCPB greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to significantly impair the capability to cool the core; and (3) GDC-31, insofar as it requires that the RCPB be designed with sufficient margin to assure that, under specified conditions, it will behave in a nonbrittle manner and the probability of a rapidly 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 conditions 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 covered (1) the description of the causes of the transient and the transient itself, (2) the initial conditions, (3) the values of 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, insofar as it requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including AOOs; (2) GDC-20, insofar as it requires that the reactor protection system be designed to initiate automatically the operation of appropriate systems, including the reactivity control systems, to ensure that SAFDLs are not exceeded as a result of AOOs; and (3) GDC-25, insofar as it requires that the protection system be designed to assure that SAFDLs are not exceeded for any 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 necessary 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 SAFDLs 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 covered (1) the description of the causes of the AOO and the description of the event itself, (2) the initial conditions, (3) the values of 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, insofar as it requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including AOOs; (2) GDC-20, insofar as it requires that the reactor protection system be designed to initiate automatically the operation of appropriate systems, including the reactivity control systems, to ensure that SAFDLs are not exceeded as a result of AOOs; and (3) GDC-25, insofar as it requires that the protection system be designed to assure that SAFDLs are not exceeded for any 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 SAFDLs 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 Control Rod Misoperation

Regulatory Evaluation

The NRC staff's review covered the types of control rod misoperations that are assumed to occur, including those caused by a system malfunction or operator error. The review covered (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, insofar as it requires that the reactor core be designed with appropriate margin to assure that SAFDLs (SAFDLs) are not exceeded during any condition of normal operation, including the effects of AOOs; (2) GDC-20, insofar as it requires that the protection system be designed to initiate the reactivity control systems automatically to assure that acceptable fuel design limits are not exceeded as a result of AOOs and to initiate automatic operation of systems and components important to safety under accident conditions; and (3) GDC-25, insofar as it requires that the protection system be designed to assure that SAFDLs are not exceeded for any 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 SAFDLs will not be exceeded during normal or anticipate operational transients. 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 covered (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, insofar as it requires that the RCS be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs; (2) GDC-20, insofar as it requires that the protection system be designed to automatically initiate the operation of appropriate systems to ensure that SAFDLs are not exceeded as a result of operational occurrences; (3) GDC-15, insofar as it requires that the RCS and its associated auxiliary systems be designed with sufficient margin to ensure that the design condition of the RCPB are not exceeded during AOOs; (4) GDC-28, insofar as it requires that the reactivity control systems be designed to assure that the effects of postulated reactivity accidents can neither result in damage to the RCPB greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to significantly impair the capability to cool the core; and (5) GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operation, including AOOs, SAFDLs are not exceeded. 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 SAFDLs 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 should stop this unplanned dilution before the shutdown margin is eliminated. The NRC staff's review covered (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, insofar as it requires that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including AOOs; (2) GDC-15, insofar as it requires that the RCS and associated auxiliary, control, and protection systems be designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOOs; and (3) GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operation, including AOOs, SAFDLs are not exceeded. 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 SAFDLs 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 covered 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, insofar as it requires that the reactivity control systems be designed to assure that the effects of postulated reactivity accidents can neither result in damage to the RCPB greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to impair significantly 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 covered (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, insofar as it requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including AOOs; (2) GDC-15, insofar as it requires that the RCS and its associated auxiliary systems be designed with sufficient margin to ensure that the design conditions of the RCPB are not exceeded during AOOs; and (3) GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operation, including AOOs, SAFDLs are not exceeded. 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 SAFDLs 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 covered (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, insofar as it requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including AOOs; (2) GDC-15, insofar as it requires that the RCS and its associated auxiliary systems be designed with sufficient margin to ensure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOOs; and (3) GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operation, including AOOs, SAFDLs are not exceeded. 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 SAFDLs 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 SG tube and main steam safety or atmospheric relief valves. Reactor protection and ESFs 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 covered (1) postulated initial core and plant conditions, (2) method of thermal and hydraulic analysis, (3) the sequence of events (assuming offsite power either available or unavailable), (4) assumed reactions of reactor system components, (5) functional and operational characteristics of the reactor protection system, (6) 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 of the SGTR is focused on the thermal and hydraulic analysis for the SGTR in order to (1) determine whether 10 CFR Part 100 is satisfied with respect to radiological consequences, which are discussed in Section 2.7 of this safety evaluation and (2) confirm that the faulted SG does not experience an overfill. Preventing SG overfill is necessary in order to prevent the failure of main steam lines. 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 faulted SG. 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

LOCAs are postulated accidents that would result in 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 to replenish it. 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 these accidents. The NRC staff's review covered (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, and calculations of the reactor power, pressure, flow, and temperature transients; (5) calculations of 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 protection and ECCS systems; and (7) operator actions. The NRC's acceptance criteria are based on (1) 10 CFR § 50.46, insofar as it establishes standards for the calculation of ECCS performance and acceptance criteria for that calculated performance; (2) 10 CFR Part 50, Appendix K, insofar as it establishes required and acceptable features of evaluation models for heat removal by the ECCS after the blowdown phase of a LOCA; (3) GDC-4, insofar as it requires that SSCs important to safety be protected against dynamic effects associated with flow instabilities and loads such as those resulting from water hammer; (4) GDC-27, insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to assure the capability to cool the core is maintained; and (5) GDC-35, insofar as it requires that a system to provide abundant emergency core cooling be provided to transfer heat from the reactor core following any LOCA at a rate so that fuel clad damage that could interfere with continued effective core cooling will be prevented. 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 was conducted to ensure that (1) the above requirements are met, and (2) 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 verified that the consequences of an ATWS are acceptable. The acceptance criterion 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 reviewed (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. *Insert the following sentence if the licensee relied upon generic vendor analyses* **[The NRC staff reviewed the licensee's justification of the applicability of generic vendor analyses 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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, as explained above, that the peak primary**

system pressure following an ATWS event will remain below the acceptance limit of 3200 psig.] 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 needs. The NRC staff's review covered the ability of the storage facilities to maintain the new fuel in a subcritical array during all credible storage conditions. The review focused 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, insofar as it requires the prevention of criticality in fuel storage systems by physical systems or processes, preferably utilizing geometrically safe configurations. Specific review criteria are contained in SRP Section 9.1.1.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 covered 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, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, and (2) GDC-62, insofar as it requires that criticality in the fuel storage systems be prevented by physical systems or processes, preferably by use of geometrically safe configurations. Specific review criteria are contained in SRP Section 9.1.2.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 3.3 - 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 reviewed the radioactive source term associated with EPUs to ensure the adequacy of the sources of radioactivity used by the licensee as input to calculations 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 included 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 radionuclides other than fission products 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, insofar as it establishes requirements for radioactivity in liquid and gaseous effluents released to unrestricted areas; (2) 10 CFR Part 50, Appendix I, insofar as it establishes numerical guides for design objectives and limiting conditions for operation to meet the "as low as is reasonably achievable" criterion; and (3) GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents. Specific review criteria are contained in SRP Section 11.1.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 reviewed 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 included (1) the sequence of events; and (2) models, assumptions, and values of parameter inputs used by the licensee for the calculation of the total effective dose equivalent (TEDE). The NRC's acceptance criteria for radiological consequences analyses using an alternate source term are based on (1) 10 CFR 50.67, insofar as it sets standards for radiological consequences of a postulated accident, and (2) GDC 19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem TEDE, as defined in 10 CFR 50.2, for the duration of the 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 reviewed the implementation of alternative source terms. The NRC's acceptance criteria for implementation of an alternative source term are based on (1) 10 CFR 50.67, insofar as it sets standards for the implementation of an alternative source term in current operating nuclear power plants; (2) 10 CFR 50.49, insofar as it requires qualification of safety-related equipment, as defined in that section, including and based on integrated radiation dose during normal and accident conditions; (3) GDC 19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem TEDE, as defined in 10 CFR 50.2, for the duration of the accident; (4) Paragraph IV.E.8 of 10 CFR Part 50, Appendix E, insofar as it requires a licensee onsite technical support center and a licensee near-site emergency operations facility from which effective direction can be given and effective control can be exercised during an emergency; and (5) 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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, as set forth above, 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. Radiological Consequences of Main Steamline Failures Outside Containment

Regulatory Evaluation

The NRC staff reviewed the analyses of the radiological consequences of a main steamline break (MSLB) outside the containment. The NRC staff's review included (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, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident, and (2) 10 CFR Part 100, insofar as it establishes requirements for assuring that radiological doses from postulated accidents will be acceptably low. 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.

Technical Evaluation

[Insert technical evaluation. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 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 reviewed the analyses of the radiological consequences of a reactor coolant pump locked-rotor accident. The review included (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, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident, and (2) 10 CFR Part 100, insofar as it establishes requirements for assuring that radiological doses from postulated accidents will be acceptably low. 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 reviewed the analyses of the radiological consequences of a control rod ejection accident. The NRC staff's review included 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 was 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, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident, and (2) 10 CFR Part 100, insofar as it establishes requirements for assuring that radiological doses from postulated accidents will be acceptably low. 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 reviewed 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 included (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, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident, and (2) GDC-55, insofar as it establishes isolation requirements for small-diameter lines connected to the primary system that form 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 reviewed the analysis of the radiological consequences of a postulated steam generator tube rupture (SGTR). The NRC staff's review included (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 included 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, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident, and (2) 10 CFR Part 100, insofar as it establishes requirements for assuring that radiological doses from postulated accidents will be acceptably low. 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 reviewed the analyses of the radiological consequences of a design-basis LOCA. The review included 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 included (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 values of 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 considered 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, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident, and (2) 10 CFR Part 100, insofar as it establishes requirements for assuring that radiological doses from postulated accidents will be acceptably low. 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 reviewed the analyses of the radiological consequences of a postulated FHA. The purpose of this review was 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 included (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, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident; (2) GDC 61, insofar as it requires that systems that contain radioactivity be designed with appropriate containment, confinement, and filtering systems; and (3) 10 CFR Part 100, insofar as it establishes requirements for assuring that radiological doses from postulated accidents will be acceptably low. 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 reviewed 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 was conducted to verify various design and operations aspects of the system. The NRC staff's review included (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 habitability. The NRC's acceptance criteria for the radiological consequences of spent fuel cask drop accidents are based on (1) GDC-19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident; (2) GDC-61, insofar as it requires that systems that contain radioactivity be designed with appropriate containment, confinement, and filtering systems; and (3) 10 CFR Part 100, insofar as it establishes requirements for assuring that radiological doses from postulated accidents will be acceptably low. 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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

FOR

SECTION 3.3 - PWR TEMPLATE SAFETY EVALUATION

2.10 Health Physics

2.10.1 Occupational and Public Radiation Doses

Regulatory Evaluation

The NRC staff conducted 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 included 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 evaluated how personnel doses needed to access plant vital areas following an accident are affected. The NRC staff considered 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 was conducted to ensure that operator performance is not adversely affected as a result of system changes made to implement the proposed EPU. The NRC staff's review covered changes to operator actions, human-system interfaces, and procedures and training needed for the proposed EPU. The NRC's acceptance criteria for human factors are based on GDC-19, 10 CFR 50.120, 10 CFR Part 55, and the guidance in 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 evaluation of the responses.

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 NRC staff statement on why the response is acceptable]

2. Changes to Operator Actions Sensitive to Power Uprate

Describe any new operator actions needed 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 involve 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 NRC staff statement on why the response is acceptable]

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 will be tested to determine they could use the instruments reliably. (SRP Section 18.0)

[Insert licensee's response followed by NRC staff statement on why the response is acceptable]

4. Changes on the Safety Parameter Display System

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

[Insert licensee's response followed by NRC staff statement on why the response is acceptable]

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

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

[Insert licensee's response followed by NRC staff statement on why the response is acceptable]

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 50.120, 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 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 NRC's acceptance criteria for the proposed EPU test program are based on 10 CFR Part 50, Appendix B, Criterion XI, which requires 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 proposed EPU, or modified to support the proposed EPU, 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 EPU

Regulatory Evaluation

The licensee conducted a risk evaluation 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 present if any issue would potentially rebut the presumption of adequate protection provided by the licensee to meet the deterministic requirements and regulations. The NRC staff’s review covered 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 covered the quality of the risk analyses used by the licensee to support the application for the proposed EPU. This included a review of the licensee’s 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. The technical evaluation should (1) clearly explain why the proposed changes satisfy each of the requirements in the regulatory evaluation and (2) provide a clear link to the conclusions reached by the NRC staff, as documented in the conclusion section.]

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 necessary 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.