OFFICE OF NUCLEAR REACTOR REGULATION

REVIEW STANDARD FOR EXTENDED POWER UPRATES

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RS-001 (DRAFT)
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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**
December 2000	Initial issuance for interim use and public comment	*Federal Register* Power Uprate Web site ADAMS	[TBD]

*Previously Concurred

CONCURRENCE					
STAFF		**PROGRAM CHAMPION**		**LEADERSHIP TEAM**	
MShuaibi/PM	12/20/02	KMcConnell	12/20/02	LMarsh for JZwolinski	12/20/02
RBouling/LA	12/20/02			BBoger*	12/18/02
LRaghavan/SC	12/20/02			GHolahan*	12/20/02
				DMatthews*	12/18/02
				RBarrett*	12/18/02

ADAMS Accession No. ML023540562

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PURPOSE

The purpose of this review standard is to provide guidance for the staff's review of extended power uprate (EPU) applications to enhance the consistency, quality, and completeness of the 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 an EPU application, licensees are encouraged to provide, with its EPU application, markups of the matrices in Section 2.1 of this review standard to identify any differences between the information in the review standard and the licensing basis of the plant.

BACKGROUND

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

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

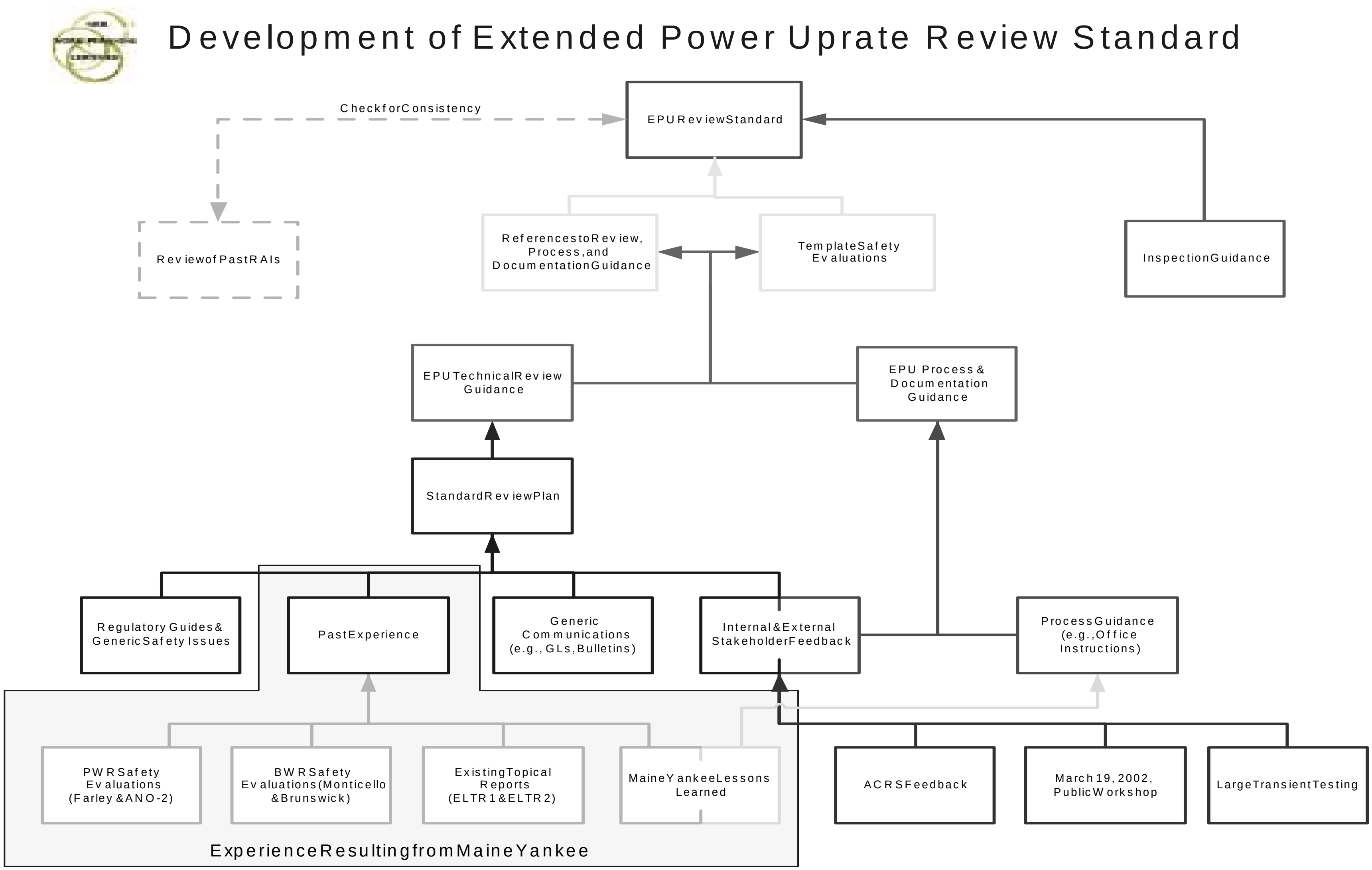
This review standard also informs licensees of the guidance documents the staff will use when reviewing EPU applications. This will help licensees prepare EPU applications that address those topics required for a complete application. By addressing the areas in the review standard, a licensee could prepare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

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

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

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

FIGURE 1


Development of Extended Power Uprate Review Standard
Check for Consistency
EPU Review Standard
Review of Past RAIs
References to Review, Process, and Documentation Guidance
Template Safety Evaluations
Inspection Guidance
EPU Technical Review Guidance
EPU Process & Documentation Guidance
Standard Review Plan
Regulatory Guides & Generic Safety Issues
Past Experience
Generic Communications (e.g., GLs, Bulletins)
Internal & External Stakeholder Feedback
Process Guidance (e.g., Office Instructions)
PWR Safety Evaluations (Farley & ANO-2)
BWR Safety Evaluations (Monticello & Brunswick)
Existing Topical Reports (ELTR 1 & ELTR 2)
Maine Yankee Lessons Learned
ACRS Feedback
March 19, 2002, Public Workshop
Large Transient Testing
Experience Resulting from Maine Yankee

GUIDANCE

This review standard provides guidance for

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

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

SECTION 1

PROCEDURAL GUIDANCE

1.1 Processing Extended Power Uprate Applications

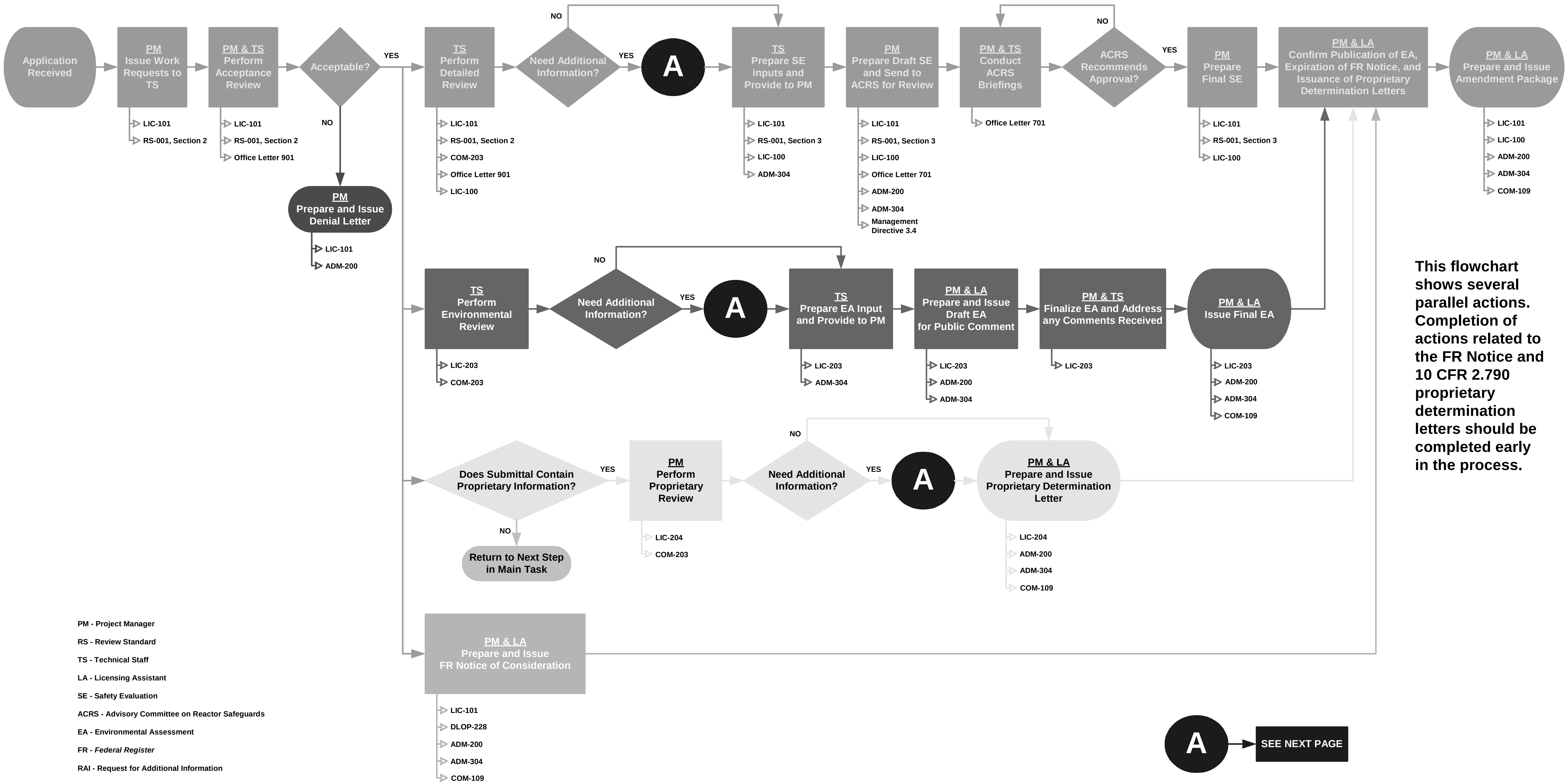
The process flow chart (Figure 1.1-1) identifies each step involved in processing an EPU application. The flow chart also identifies the responsible individual/organization and applicable procedures for completing each step. The staff should use the flow chart and referenced guidance documents when processing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

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

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

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

Figure 1.1-1 EPU Process Flow Chart

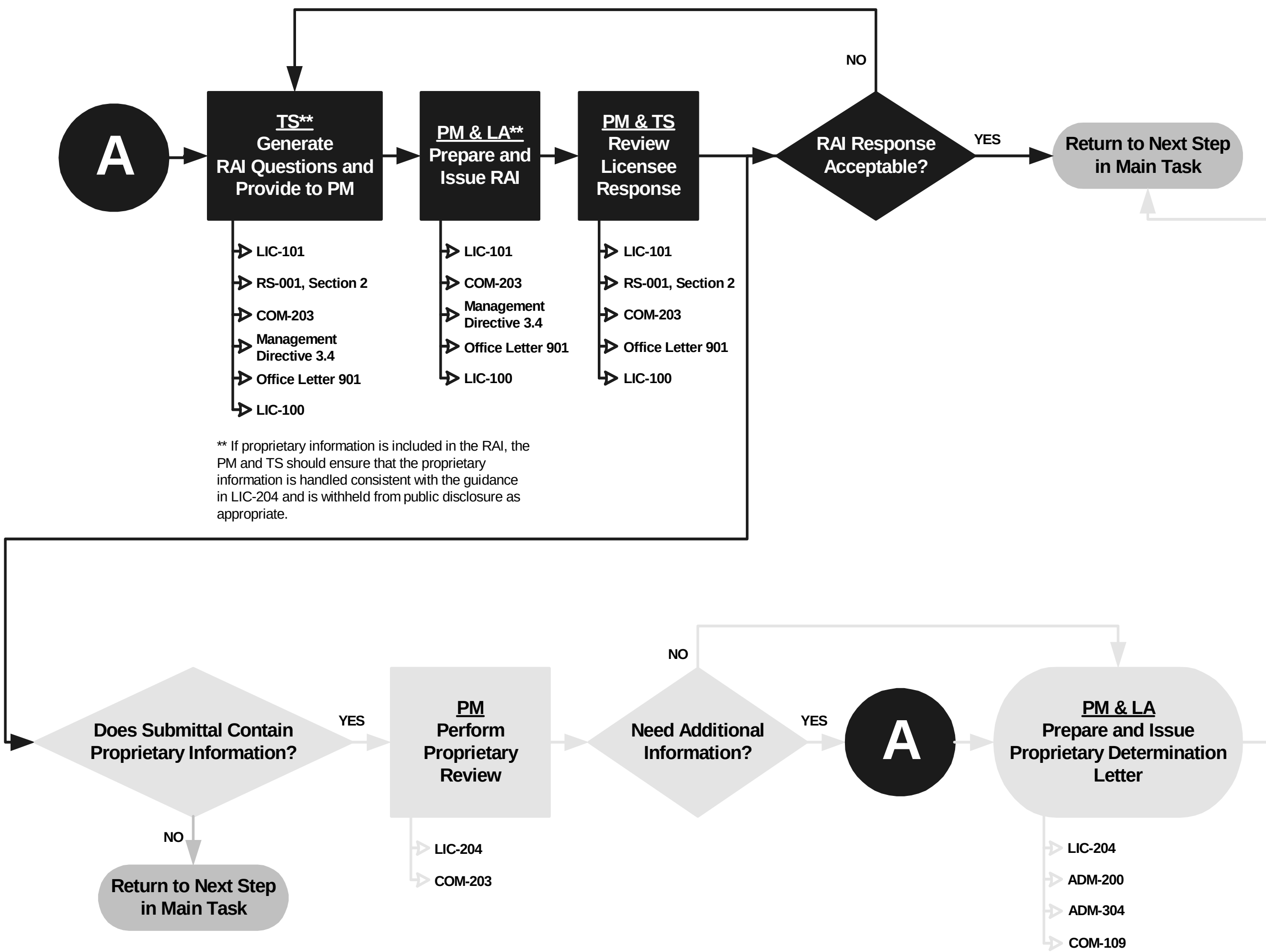


This flowchart shows several parallel actions. Completion of actions related to the FR Notice and 10 CFR 2.790 proprietary determination letters should be completed early in the process.

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

Figure 1.1-1 EPU Process Flow Chart

continued



PM - Project Manager

RS - Review Standard

TS - Technical Staff

LA - Licensing Assistant

SE - Safety Evaluation

ACRS - Advisory Committee on Reactor Safeguards

EA - Environmental Assessment

FR - *Federal Register*

RAI - Request for Additional Information

SECTION 2

TECHNICAL REVIEW GUIDANCE

2.1 Reviewing Extended Power Uprate Applications

This section defines the scope of technical review for EPU applications and identifies the guidance to be used when performing the technical review 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 review are included in the referenced guidance documents.

The review involves the following three steps:

Step 1. Initial Screening

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

Step 2. Acceptance Review

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

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

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

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

(3) Use the “Acceptance Review” column of the matrices as a checklist to document whether the licensee has addressed the areas of review in sufficient detail to allow the staff to proceed with its detailed 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 review, provide the plant Project Manager a copy of the matrix completed as a result of instruction (3) above.

Step 3. Detailed Technical Review

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

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

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

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

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

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

(7) Document the results of the 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
							BWR	PWR	
Reactor Vessel Material Surveillance Program	All EPUs	EMCB	SRXB	5.3.1 Draft Rev. 2 April 1996	GDC-14 GDC-31 10 CFR 50, App. H 10 CFR 50.60	RG 1.190	2.1.1	2.1.1	
Pressure-Temperature Limits and Upper-Shelf Energy	All EPUs	EMCB	SRXB	5.3.2 Draft Rev. 2 April 1996	GDC-14 GDC-31 10 CFR 50, App. G 10 CFR 50.60	RG 1.161 RG 1.190 RG 1.99	2.1.2	2.1.2	
Pressurized Thermal Shock	PWR EPUs	EMCB	SRXB	5.3.2 Draft Rev. 2 April 1996	GDC-14 GDC-31 10 CFR 50.61	RG 1.190 RG 1.154		2.1.3	
Reactor Internal and Core Support Materials	All EPUs	EMCB	SRXB	4.5.2 Draft Rev. 3 April 1996	GDC-1 10 CFR 50.55a	Note 1*	2.1.3	2.1.4	

<table>
<tr><th rowspan="2">Areas of Review</th><th rowspan="2">Applicable to</th><th rowspan="2">Primary Review Branch</th><th rowspan="2">Secondary Review Branch(es)</th><th rowspan="2">SRP Section Number</th><th rowspan="2">Focus of SRP Usage</th><th rowspan="2">Other Guidance</th><th colspan="2">Template Safety Evaluation Section Number</th><th rowspan="2">Acceptance Review</th></tr>
<tr><th>BWR</th><th>PWR</th></tr>
<tr><td rowspan="6">Reactor Coolant Pressure Boundary Materials</td><td rowspan="6">All EPUs</td><td rowspan="6">EMCB</td><td rowspan="6">EMEB
SRXB</td><td>5.2.3
Draft Rev. 3
April 1996</td><td>GDC-1
10 CFR 50.55a
GDC-4
GDC-14
GDC-31
10 CFR 50, App. G</td><td rowspan="6">RG 1.190
GL 97-01
IN 00-17s1
BL 01-01
BL 02-01
BL 02-02
Note 2*
Note 3*</td><td rowspan="6">2.1.4</td><td rowspan="6">2.1.5</td><td rowspan="6"></td></tr>
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April 1996</td><td>GDC-1
10 CFR 50.55a
GDC-14</td></tr>
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GDC-4
GDC-14
GDC-31
10 CFR 50, App. G</td></tr>
<tr><td>5.3.3
Draft Rev. 2
April 1996</td></tr>
<tr><td>6.1.1
Draft Rev. 2
April 1996</td></tr>
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Draft
Aug. 1987</td><td>GDC-4</td><td>NUREG
1061
Vol. 3
Nov. 1984</td><td></td><td>2.1.6</td><td></td></tr>
<tr><td>Protective Coating Systems (Paints) - Organic Materials</td><td>All EPUs</td><td>EMCB</td><td></td><td>6.1.2
Draft Rev. 3
April 1996</td><td>10 CFR 50, App. B
RG 1.54</td><td></td><td>2.1.5</td><td>2.1.7</td><td></td></tr>
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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
							BWR	PWR	
Effect of EPU on Flow-Accelerated Corrosion	All EPUs	EMCB				Note 4*	2.1.6	2.1.8	
Steam Generator Tube Inservice Inspection	PWR EPUs	EMCB		5.4.2.2 Draft Rev. 2 April 1996	10 CFR 50.55a	Plant TSs RG 1.121 GL 95-03 BL 88-02 GL 95-05 Note 5*		2.1.9	
Steam Generator Blowdown System	PWR EPUs	EMCB		10.4.8 Draft Rev. 3 April 1996	GDC-14			2.1.10	
Chemical and Volume Control System (Including Boron Recovery System)	PWR EPUs	EMCB	SPLB SRXB	9.3.4 Draft Rev. 3 April 1996	GDC-14 GDC-29			2.1.11	
Reactor Water Cleanup System	BWR EPUs	EMCB		5.4.8 Draft Rev. 3 April 1996	GDC-14 GDC-60 GDC-61		2.1.7		

Notes:

1. In addition to the SRP, guidance on 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, applicants 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, Rev. 2, 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, April 1999, "Recommendations for Effective an Flow-Accelerated Corrosion Program." This EPRI document is copyrighted. EPRI has provided copies of this document to EMCB for use by NRC staff. Copying of this document, however, is not allowed.

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

LIST OF ACRONYMS FOR MATRIX 1

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

ATTACHMENT 1 TO MATRIX 1

Independent Calculations

Materials and Chemical Engineering

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

MATRIX 2

SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

Mechanical and Civil Engineering

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

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

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

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

LIST OF ACRONYMS FOR MATRIX 2

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

ATTACHMENT 1 TO MATRIX 2

Independent Calculations

Mechanical and Civil Engineering

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

MATRIX 3

SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

Electrical Engineering

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

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

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

LIST OF ACRONYMS FOR MATRIX 3

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

ATTACHMENT 1 TO MATRIX 3

Independent Calculations

Electrical Engineering

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

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

MATRIX 4

SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

Instrumentation and Controls

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

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

LIST OF ACRONYMS FOR MATRIX 4

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

ATTACHMENT 1 TO MATRIX 4

Independent Calculations

Instrumentation and Controls

Independent calculations are not performed in the area of instrumentation and controls. For a plant where an instrument setpoint methodology has not been previously approved, a detailed review of the licensee's calculations for one instrument should be performed to verify proper application of the methodology.

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
							BWR	PWR	
Flood Protection	EPUs that result in significant increases in fluid volumes of tanks and vessels	SPLB		3.4.1 Rev. 2 July 1981	GDC-2		2.5.1.1.1	2.5.1.1.1	
Equipment and Floor Drainage System	EPUs that result in increases in fluid volumes or in installation of larger capacity pumps or piping systems	SPLB		9.3.3 Rev. 2 July 1981	GDC-2 GDC-4		2.5.1.1.2	2.5.1.1.2	
Circulating Water System	EPUs that result in increases in fluid volumes associated with the circulating water system or in installation of larger capacity pumps or piping systems	SPLB		10.4.5 Rev. 2 July 1981	GDC-4		2.5.1.1.3	2.5.1.1.3	
Internally Generated Missiles (Outside Containment)	EPUs that result in substantially higher system pressures or changes in existing system configuration	SPLB	EMCB EMEB	3.5.1.1 Rev. 2 July 1981	GDC-4		2.5.1.2.1	2.5.1.2.1	
Internally Generated Missiles (Inside Containment)	EPUs that result in substantially higher system pressures or changes in existing system configuration	SPLB	EMCB EMEB	3.5.1.2 Rev. 2 July 1981	GDC-4		2.5.1.2.1	2.5.1.2.1	

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review
							BWR	PWR	
Turbine Generator	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB		10.2 Rev. 2 July 1981	GDC-4		2.5.1.2.2	2.5.1.2.2	
Protection Against Postulated Piping Failures in Fluid Systems Outside Containment	EPUs that affect environmental conditions, habitability of the control room, or access to areas important to safe control of postaccident operations	SPLB	EMCB EMEB	3.6.1 Rev. 1 July 1981	GDC-4		2.5.1.3	2.5.1.3	
Fire Protection Program	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB		9.5.1 Rev. 3 July 1981	10 CFR 50.48 10 CFR 50, App. R GDC-3 GDC-5	Note 1*	2.5.1.4	2.5.1.4	
PWR Dry Containments, Including Subatmospheric Containments	EPUs for PWR plants with dry containments (including subatmospheric containments) except where the application demonstrates that previous analysis is bounding	SPLB		6.2.1 Rev. 2 July 1981 6.2.1.1.A Rev. 2 July 1981	GDC-13 GDC-16 GDC-38 GDC-50 GDC-64			2.5.2.1	
Ice Condenser Containments	EPUs for PWR plants with ice condenser containments except where the application demonstrates that previous analysis is bounding	SPLB		6.2.1 Rev. 2 July 1981 6.2.1.1.B Rev. 2 July 1981	GDC-13 GDC-16 GDC-38 GDC-50 GDC-64			2.5.2.1	

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
							BWR	PWR	
Pressure-Suppression Type BWR Containments	EPUs for BWR plants with pressure-suppression containments except where the application demonstrates that previous analysis is bounding	SPLB		6.2.1 Rev. 2 July 1981 6.2.1.1.C Rev. 6 Aug. 1984	GDC-4 GDC-13 GDC-16 GDC-50 GDC-64		2.5.2.1		
Subcompartment Analysis	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB		6.2.1 Rev. 2 July 1981 6.2.1.2 Rev. 2 July 1981	GDC-4 GDC-50		2.5.2.2	2.5.2.2	
Mass and Energy Release Analysis for Postulated Loss-of-Coolant	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB		6.2.1 Rev. 2 July 1981 6.2.1.3 Rev. 1 July 1981	GDC-50 10 CFR 50, App. K		2.5.2.3.1	2.5.2.3.1	
Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures	PWR EPUs except where the application demonstrates that previous analysis is bounding	SPLB		6.2.1 Rev. 2 July 1981 6.2.1.4 Rev. 1 July 1981	GDC-50			2.5.2.3.2	

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review
							BWR	PWR	
Combustible Gas Control In Containment	EPUs that impact hydrogen release assumptions	SPLB		6.2.5 Rev. 2 July 1981	10 CFR 50.44 10 CFR 50.46 GDC-5 GDC-41 GDC-42 GDC-43		2.5.2.4	2.5.2.4	
Containment Heat Removal	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB		6.2.2 Rev. 4 Oct. 1985	GDC-38	DG-1107	2.5.2.5	2.5.2.5	
Secondary Containment Functional Design	EPUs that affect the pressure and temperature response, or draw-down time of the secondary containment	SPLB		6.2.3 Rev. 2 July 1981	GDC-4 GDC-16		2.5.2.6		
Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies	PWR EPUs except where the application demonstrates that previous analysis is bounding	SPLB	SRXB	6.2.1 Rev. 2 July 1981 6.2.1.5 Rev. 2 July 1981	10 CFR 50.46 10 CFR 50, App. K			2.5.2.6	
Pressurizer Relief Tank	PWR EPUs that affect pressurizer discharge to the PRT	SPLB	EMEB	5.4.11 Rev. 2 July 1981	GDC-2 GDC-4			2.5.2.7	
Control Room Habitability System	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB	SPSB	6.4 Draft Rev. 3 April 1996	GDC-4 GDC-19	Note 2* Note 3*	2.5.3.1	2.5.3.1	
ESF Atmosphere Cleanup System	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB	SPSB	6.5.1 Rev. 2 July 1981	GDC-19 GDC-41 GDC-61 GDC-64		2.5.3.2	2.5.3.2	

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review
							BWR	PWR	
Fission Product Control Systems and Structures	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB	EMCB	6.5.3 Rev. 2 July 1981	GDC-41		2.5.3.3	2.5.3.3	
Main Condenser Evacuation System	EPUs for which the main condenser evacuation system is modified	SPLB		10.4.2 Rev. 2 July 1981	GDC-60 GDC-64		2.5.3.4	2.5.3.4	
Turbine Gland Sealing System	EPUs for which the turbine gland sealing system is modified	SPLB		10.4.3 Rev. 2 July 1981	GDC-60 GDC-64		2.5.3.5	2.5.3.5	
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		2.5.3.6		
Control Room Area Ventilation System	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB	SPSB	9.4.1 Rev. 2 July 1981	GDC-4 GDC-19 GDC-60		2.5.4.1	2.5.4.1	
Spent Fuel Pool Area Ventilation System	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB	SPSB	9.4.2 Rev. 2 July 1981	GDC-60 GDC-61		2.5.4.2	2.5.4.2	
Auxiliary and Radwaste Area Ventilation System	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB		9.4.3 Rev. 2 July 1981	GDC-60		2.5.4.3	2.5.4.3	
Turbine Area Ventilation System	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB		9.4.4 Rev. 2 July 1981	GDC-60		2.5.4.3	2.5.4.3	
ESF Ventilation System	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB		9.4.5 Rev. 2 July 1981	GDC-4 GDC-17 GDC-60		2.5.4.4	2.5.4.4	

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
							BWR	PWR	
Spent Fuel Pool Cooling and Cleanup System	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB	EMCB	9.1.3 Rev. 1 July 1981	GDC-5 GDC-44 GDC-61	Note 4*	2.5.5.1	2.5.5.1	
Station Service Water System	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB		9.2.1 Rev. 4 June 1985	GDC-4 GDC-5 GDC-44	GL 89-13 and Suppl. 1 GL 96-06 and Suppl. 1	2.5.5.2	2.5.5.2	
Reactor Auxiliary Cooling Water Systems	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB		9.2.2 Rev. 3 June 1986	GDC-4 GDC-5 GDC-44	GL 89-13 and Suppl. 1 GL 96-06 and Suppl. 1	2.5.5.3	2.5.5.3	
Ultimate Heat Sink	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB		9.2.5 Rev. 2 July 1981	GDC-5 GDC-44		2.5.5.4	2.5.5.4	
Auxiliary Feedwater System	PWR EPUs except where the application demonstrates that previous analysis is bounding	SPLB		10.4.9 Rev. 2 July 1981	GDC-4 GDC-5 GDC-19 GDC-34 GDC-44			2.5.5.5	
Main Steam Supply System	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB		10.3 Rev. 3 April 1984	GDC-4 GDC-5 GDC-34		2.5.6.1	2.5.6.1	

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review
							BWR	PWR	
Main Condenser	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB		10.4.1 Rev. 2 July 1981	GDC-60		2.5.6.2	2.5.6.2	
Turbine Bypass System	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB		10.4.4 Rev. 2 July 1981	GDC-4 GDC-34		2.5.6.3	2.5.6.3	
Condensate and Feedwater System	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB		10.4.7 Rev. 3 April 1984	GDC-4 GDC-5 GDC-44		2.5.6.4	2.5.6.4	
Gaseous Waste Management Systems	EPUs that impact the level of fission products in the reactor coolant system, or the amount of gaseous waste	SPLB	IEHB	11.3 Draft Rev. 3 April 1996	10 CFR 20.1302 GDC-3 GDC-60 GDC-61 10 CFR 50, App. I		2.5.7.1	2.5.7.1	
Liquid Waste Management Systems	EPUs that impact the level of fission products in the reactor coolant system, or the amount of liquid waste	SPLB	IEHB	11.2 Draft Rev. 3 April 1996	10 CFR 20.1302 GDC-60 GDC-61 10 CFR 50, App. I		2.5.7.2	2.5.7.2	
Solid Waste Management Systems	EPUs that impact the level of fission products in the reactor coolant system, or the amount of solid waste	SPLB	IEHB	11.4 Draft Rev. 3 April 1996	10 CFR 20.1302 GDC-60 GDC-63 GDC-64 10 CFR 71		2.5.7.3	2.5.7.3	
Emergency Diesel Engine Fuel Oil Storage and Transfer System	EPUs that result in higher EDG electrical demands	SPLB		9.5.4 Rev. 2 July 1981	GDC-4 GDC-5 GDC-17		2.5.8.1	2.5.8.1	
Light Load Handling System (Related to Refueling)	EPUs except where the application demonstrates that previous analysis is bounding	SPLB	SPSB	9.1.4 Rev. 2 July 1981	GDC-61 GDC-62		2.5.8.2	2.5.8.2	

Notes:

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

2. 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.”

3. 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.

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

LIST OF ACRONYMS FOR MATRIX 5

BWR = boiling-water reactor
CFR = *Code of Federal Regulations*
DG = draft guide
EMCB = Materials & Chemical Engineering Branch
EMEB = Mechanical & Civil Engineering Branch
EPUs = extended power uprates
ESF = engineered safety feature
GDC = General Design Criterion
GL = generic letter
IEHB = Equipment and Human Performance Branch
PWR = pressurized-water reactor
SPLB = Plant Systems Branch
SPSB = Probabalistic Safety Assessment Branch
SRP = Standard Review Plan
SRXB = Reactor Systems Branch

ATTACHMENT 1 TO MATRIX 5

Independent Calculations

Plant Systems

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

ATTACHMENT 2 TO MATRIX 5

Supplemental Fire Protection Review Criteria

Plant Systems

This attachment provides guidance for the review of the fire protection information to be provided in an application for a power uprate. Power uprates typically result in an increase in decay heat generation following a plant trip. This increase in decay heat usually does 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, the 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 for a fire event. In this case, a specific analysis for fire events may not be required. However, where licensees rely on less than full capability systems for fire events (e.g., partial automatic depressurization system capability for reduced capability makeup pump) the licensee should provide specific analyses for fire events that demonstrate that (1) fuel integrity is maintained by demonstrating that the fuel design limits are not exceeded and (2) there are no adverse consequences on the reactor pressure vessel integrity or the attached piping. Plants that rely on alternative/dedicated or backup shutdown capability for post-fire safe shutdown should analyze the impact of the power uprate on the alternative/dedicated or backup shutdown capability. The staff should verify that the capability of the alternative/dedicated or backup systems relied upon for post-fire safe shutdown are capable of achieving and maintaining safe shutdown considering the impact of the power uprate.

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

ATTACHMENT 3 TO MATRIX 5

Supplemental Spent Fuel Pool Cooling Review Criteria

Plant Systems

1. BACKGROUND

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

A significant design-basis challenge to the SFP cooling system is imposed by a planned evolution (fuel transfer from the reactor vessel). Emergency offloads are not considered credible because fuel transfers require plant cooldown, reactor disassembly, and refueling cavity flooding, which are time-consuming, manual processes. As a result, 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) will be reviewed.

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 fuel pool surface is at a sufficiently low rate to preclude fogging, and (3) SFP temperature is within the design range of the cleanup system demineralizes to maintain water clarity. Defense-in-depth is provided by:

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

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

3. REVIEW PROCEDURES

3.1. Adequate SFP Cooling Capacity

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

3.1.1. Bounding Calculation

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

3.1.1.1. Full Cooling System Capability Evaluation

Analysis conditions:

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

3.1.1.2. Single-Active Failure Evaluation

Analysis conditions:

(1) decay heat load is calculated based on a bounding estimate of offload size, decay time, power history, and inventory of previously discharged assemblies
(2) heat removal capability is based on a bounding estimate of ultimate heat sink temperature, heat exchanger performance (e.g., fouling and tube plugging margin), and cooling system flow rates assuming the limiting single failure with regard to heat removal capability
(3) alternate heat removal paths (e.g., evaporative cooling) must be appropriately validated and based on bounding input parameter values (e.g., air temperature, relative humidity, and ventilation flow rate)
(4) calculated bulk SFP temperature must remain below the design temperature of the SFP structure and liner, and calculated peak storage cell temperature must remain below the storage rack design temperature
(5) for plants where a single failure results in a complete loss of forced cooling, the 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 to support operational flexibility.

3.1.2. Cycle-Specific Calculation:

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

Cycle-specific analysis conditions:

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

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

3.2. Adequate Make-Up Supply

(1) Following a loss-of-SFP cooling event, the licensee must be able to provide two sources of make-up water prior to the occurrence of boiling in the pool. To determine the time to boil, the initial pool temperature is the peak temperature from a planned offload, assuming the worst single-active failure occurred.
(2) At least one make-up source shall have a capacity that is equal to or greater than the calculated boil-off rate so that the SFP level can be maintained.

MATRIX 6

SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

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
							BWR	PWR	
Fuel System Design	All EPUs	SRXB		4.2 Draft Rev. 3 April 1996	10 CFR 50.46 GDC-10 GDC-27 GDC-35	Note 1* Note 2*	2.6.1	2.6.1	
Nuclear Design	All EPUs	SRXB		4.3 Draft Rev. 3 April 1996	GDC-10 GDC-11 GDC-12 GDC-13 GDC-20 GDC-25 GDC-26 GDC-27 GDC-28	RG 1.190 GSI 170 IN 97-085	2.6.2	2.6.2	
Thermal and Hydraulic Design	All EPUs	SRXB		4.4 Draft Rev. 2 April 1996	GDC-10 GDC-12	Note 3*	2.6.3	2.6.3	

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review
							BWR	PWR	
Functional Design of Control Rod Drive System	All EPUs	SRXB	SPLB	4.6 Draft Rev. 2 April 1996	GDC-4 GDC-23 GDC-25 GDC-26 GDC-27 GDC-28 GDC-29 10 CFR 50.62(c)(3)		2.6.4.1	2.6.4.1	
Overpressure Protection during Power Operation	All EPUs	SRXB		5.2.2 Draft Rev. 3 April 1996	GDC-15 GDC-31	Note 4*	2.6.4.2	2.6.4.2	
Overpressure Protection during Low Temperature Operation	PWR EPUs	SRXB		5.2.2 Draft Rev. 3 April 1996	GDC-15 GDC-31			2.6.4.3	
Reactor Core Isolation Cooling System	BWR EPUs	SRXB		5.4.6 Draft Rev. 4 April 1996	GDC-4 GDC-5 GDC-29 GDC-33 GDC-34 GDC-54 10 CFR 50.63		2.6.4.3		
Residual Heat Removal System	All EPUs	SRXB		5.4.7 Draft Rev. 4 April 1996	GDC-4 GDC-5 GDC-19 GDC-34	Note 5*	2.6.4.4	2.6.4.4	
Emergency Core Cooling System	All EPUs	SRXB		6.3 Draft Rev. 3 April 1996	GDC-4 GDC-27 GDC-35 10 CFR 50.46 10 CFR 50 App. K	Note 6*	2.6.5.6.2	2.6.5.6.3	

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review
							BWR	PWR	
Standby Liquid Control System	BWR EPUs	SRXB	EMCB SPLB	9.3.5 Draft Rev. 3 April 1996	GDC-26 GDC-27 10 CFR 50.62(c)(4)	Note 12*	2.6.4.5		
Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve	All EPUs	SRXB		15.1.1-4 Draft Rev. 2 April 1996	GDC-10 GDC-15 GDC-20 GDC-26	Note 7*	2.6.5.1	2.6.5.1.1	
Steam System Piping Failures Inside and Outside of Containment	PWR EPUs	SRXB		15.1.5 Draft Rev. 3 April 1996	GDC-27 GDC-28 GDC-31 GDC-35	Note 7*		2.6.5.1.2	
Loss of External Load; Turbine Trip, Loss of Condenser Vacuum; Closure of Main Steam Isolation Valve (BWR); and Steam Pressure Regulator Failure (Closed)	All EPUs	SRXB		15.2.1-5 Draft Rev. 2 April 1996	GDC-10 GDC-15 GDC-26	Note 7*	2.6.5.2.1	2.6.5.2.1	
Loss of Nonemergency AC Power to the Station Auxiliaries	All EPUs	SRXB		15.2.6 Draft Rev. 2 April 1996	GDC-10 GDC-15 GDC-26	Note 7*	2.6.5.2.2	2.6.5.2.2	
Loss of Normal Feedwater Flow	All EPUs	SRXB	EEIB	15.2.7 Draft Rev. 2 April 1996	GDC-10 GDC-15 GDC-26	Note 7*	2.6.5.2.3	2.6.5.2.3	
Feedwater System Pipe Breaks Inside and Outside Containment	PWR EPUs	SRXB	EEIB	15.2.8 Draft Rev. 2 April 1996	GDC-27 GDC-28 GDC-31 GDC-35	Note 7*		2.6.5.2.4	

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
							BWR	PWR	
Loss of Forced Reactor Coolant Flow Including Trip of Pump Motor and Flow Controller Malfunctions	All EPUs	SRXB		15.3.1-2 Draft Rev. 2 April 1996	GDC-10 GDC-15 GDC-26	Note 7*	2.6.5.3.1	2.6.5.3.1	
Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break	All EPUs	SRXB		15.3.3-4 Draft Rev. 3 April 1996	GDC-27 GDC-28 GDC-31	Note 7*	2.6.5.3.2	2.6.5.3.2	
Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition	All EPUs	SRXB		15.4.1 Draft Rev. 3 April 1996	GDC-10 GDC-20 GDC-25	Note 7*	2.6.5.4.1	2.6.5.4.1	
Uncontrolled Control Rod Assembly Withdrawal at Power	All EPUs	SRXB		15.4.2 Draft Rev. 3 April 1996	GDC-10 GDC-20 GDC-25	Note 7*	2.6.5.4.2	2.6.5.4.2	
Control Rod Misoperation (System Malfunction or Operator Error)	PWR EPUs	SRXB		15.4.3 Draft Rev. 3 April 1996	GDC-10 GDC-20 GDC-25	Note 7*		2.6.5.4.3	
Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate	All EPUs	SRXB		15.4.4-5 Draft Rev. 2 April 1996	GDC-10 GDC-15 GDC-20 GDC-26 GDC-28	Note 7*	2.6.5.4.3	2.6.5.4.4	
Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant	PWR EPUs	SRXB		15.4.6 Draft Rev. 2 April 1996	GDC-10 GDC-15 GDC-26	Note 7*		2.6.5.4.5	

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review
							BWR	PWR	
Spectrum of Rod Ejection Accidents	PWR EPUs	SRXB		15.4.8 Draft Rev. 3 April 1996	GDC-28	Note 7*		2.6.5.4.6	
Spectrum of Rod Drop Accidents	BWR EPUs	SRXB		15.4.9 Draft Rev. 3 April 1996	GDC-28	Note 7*	2.6.5.4.4		
Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory	All EPUs	SRXB		15.5.1-2 Draft Rev. 2 April 1996	GDC-10 GDC-15 GDC-26	Note 7* Note 8*	2.6.5.5	2.6.5.5	
Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve or a BWR Pressure Relief Valve	All EPUs	SRXB		15.6.1 Draft Rev. 2 April 1996	GDC-10 GDC-15 GDC-26	Note 7*	2.6.5.6.1	2.6.5.6.1	
Steam Generator Tube Rupture	PWR EPUs	SRXB		15.6.3 Draft Rev. 3 April 1996	Note 7* Note 9*	Note 7* Note 9*		2.6.5.6.2	
Loss-of Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary	All EPUs	SRXB		15.6.5 Draft Rev. 3 April 1996	GDC-35 10 CFR 50.46	Note 7* Note 10*	2.6.5.6.2	2.6.5.6.3	
Anticipated Transient Without Scram	All EPUs	SRXB				Note 7* Note 11* Note 12*	2.6.5.7	2.6.5.7	
New Fuel Storage	EPU applications that request approval for new fuel.	SRXB		9.1.1 Draft Rev. 3 April 1996	GDC-62		2.6.6.1	2.6.6.1	

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review
							BWR	PWR	
Spent Fuel Storage	EPU applications that request approval for new fuel.	SRXB		9.1.2 Draft Rev. 4 April 1996	GDC-4 GDC-62		2.6.6.2	2.6.6.2	

Notes:

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

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

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

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

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

6. The review for BWRs will cover (1) the basis for use of the ISCOR computer code in emergency core cooling system analyses, (2) the spectrum of breaks analyzed, (3) justification for changes in calculated peak cladding temperature (PCT) for the licensing-basis case and the upper-bound case and any impact of the changes in PCTs on the use the licensing methods for the power uprates.

7. The review also confirms:
   - The licensee used codes and methods approved for the plant-specific application and the licensee's use of the codes and methods complies with any limitations, restrictions, and conditions specified in the approving safety evaluation.
   - All changes of reactor protection system trip delays are correctly addressed and accounted for in the analyses.
   - (For PWRs) Steam generator plugging and asymmetry limits are accounted for in the analyses.
   - (For PWRs), Any observed hot-leg streaming effects are accounted for in the analyses.
   - (For PWRs), The licensee's evaluation of the effects of Westinghouse Nuclear Service Advisory Letters (NSALs), NSAL 02-3 and Revision 1, NSAL 02-4, and NSAL 02-5. These NSALs document problems with water level setpoint uncertainties in Westinghouse-designed steam generators. The review is conducted to ensure that the effects of the problems identified have been accounted for in steam generator water level setpoints used in LOCA, non-LOCA, and ATWS analyses.

8. The following additional guidance is provided for the inadvertent operation of emergency core cooling system and chemical and volume control system malfunctions that increases reactor coolant inventory events: (a) non-safety-grade pressure-operated relief valves should not be credited for event mitigation and (b) pressurizer level should not be allowed to reach a pressurizer water-solid condition.

9. The review is also performed to confirm that steam generators do not experience an overfill in order to avoid potential equipment damage.

10. 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 licensee's analyses meet the requirements of Item II.K.3.5 of NUREG-0737 related to RCP trip analyses for small-break LOCAs
   - (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.

11. For PWRs, the ATWS review is conducted to ensure that the plant meets the following 10 CFR 50.62 requirements:
   - 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.

   - Each PWR manufactured by Combustion Engineering or Babcock and Wilcox 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 review also covers the bases for setpoints for the AMSAC and DSS to ensure that the setpoints remain valid for the uprated power level. In addition, for 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. The review covers the limiting event determination, the sequence of events, the analytical model and its applicability, the values of parameters used in the analytical model, and the results of the analyses. If the licensee relies upon generic vendor analyses, the review covers the licensee's justification of the applicability of that analysis to its plant and the operating conditions for the proposed EPU.

12. For BWR plants, the review is conducted to ensure that the licensee has appropriately accounted for changes in analysis due to the uprated power level and confirm that requried equipment, such as the standby liquid control system (SLCS) pumps can deliver required flowrates. The review will also cover the SLCS relief valve margin consistent with the experience described in Information Notice 2001-13 and any plans by the licensee in relation to this issue. 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 6

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

ATTACHMENT 1 TO MATRIX 6

Independent Calculations

Reactor Systems

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

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

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

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

MATRIX 7

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
							BWR	PWR	
Source Terms for Input into Radwaste Management Systems Analyses	All EPUs	SPSB		11.1 Draft Rev. 3 April 1996	10 CFR 20 10 CFR 50, App. I GDC-60		2.7.1	2.7.1	
Radiological Consequence Analyses Using Alternative Source Terms	EPUs that utilize alternative source term	SPSB	EEIB EMCB EMEB IEHB SPLB SRXB	15.0.1 Rev. 0 July 2000	10 CFR 50.67 GDC-19 10 CFR 50.49 10 CFR 51 10 CFR 50, App. E NUREG-0737		2.7.2	2.7.2	
Radiological Consequences of Main Steamline Failures Outside Containment for a PWR	PWR EPUs that do not utilize alternative source term whose main steamline break analyses result in fuel failure	SPSB	SRXB	15.1.5, App. A Draft Rev. 3 April 1996	10 CFR 100	Notes 4, 5, 6, 7, 27*		2.7.2	
				6.4 Draft Rev. 3 April 1996	GDC-19	Notes 1, 2, 3, 28, 29*			
Radiological Consequences of Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break	EPUs that do not utilize alternative source term whose reactor coolant pump rotor seizure or reactor coolant pump shaft break results in fuel failure	SPSB	SRXB	15.3.3-4 Draft Rev. 3 April 1996	10 CFR 100	Notes 5, 8, 9, 27*		2.7.3	
				6.4 Draft Rev. 3 April 1996	GDC-19	Notes 1, 2, 3, 28, 29*			

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Acceptance Review
							BWR	PWR	
Radiological Consequences of a Control Rod Ejection Accident	PWR EPUs that do not utilize alternative source term whose rod ejection accident results in fuel failure or melting	SPSB	SRXB	15.4.8, App. A Draft Rev. 2 April 1996	10 CFR 100	Notes 4, 21, 22, 27*		2.7.4	
				6.4 Draft Rev. 3 April 1996	GDC-19	Notes 1, 2, 3, 28, 29*			
Radiological Consequences of Control Rod Drop Accident	BWR EPUs that do not utilize alternative source term whose control rod drop accident results in fuel failure or melting	SPSB	SRXB	15.4.9, App. A Draft Rev. 3 April 1996	10 CFR 100	Notes 9, 10, 27*	2.7.2		
				6.4 Draft Rev. 3 April 1996	GDC-19	Notes 1, 2, 3, 28, 29*			
Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment	EPUs that do not utilize alternative source term whose failure of small lines carrying primary coolant outside containment result in fuel failure	SPSB		15.6.2 Draft Rev. 3 April 1996	GDC-55 10 CFR 100		2.7.3	2.7.5	
				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
							BWR	PWR	
Radiological Consequences of Steam Generator Tube Failure	PWR EPUs that do not utilize alternative source term whose steam generator tube failure results in fuel failure	SPSB	SRXB	15.6.3 Draft Rev. 3 April 1996	10 CFR 100	Notes 4, 13, 14, 15, 27*		2.7.6	
				6.4 Draft Rev. 3 April 1996	GDC-19	Notes 1, 2, 3, 28, 29*			
Radiological Consequences of Main Steamline Failure Outside Containment for a BWR	BWR EPUs that do not utilize alternative source term whose main steam line failure outside containment results in fuel failure	SPSB	SRXB	15.6.4 Draft Rev. 3 April 1996	10 CFR 100	Note 27*	2.7.4		
				6.4 Draft Rev. 3 April 1996	GDC-19	Notes 1, 2, 3, 28, 29*			
Radiological Consequences of a Design Basis Loss-Of-Coolant-Accident Including Containment Leakage Contribution	EPUs that do not utilize alternative source term	SPSB	SPLB	15.6.5, App. A Draft Rev. 2 April 1996	10 CFR 100	Notes 4, 23, 24, 25, 26, 27*	2.7.5	2.7.7	
				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
							BWR	PWR	
Radiological Consequences of a Design Basis Loss-Of-Coolant-Accident: Leakage from ESF Components Outside Containment	EPUs that do not utilize alternative source term	SPSB	SPLB	15.6.5, App. B Draft Rev. 2 April 1996	10 CFR 100	Notes 11, 27*	2.7.5	2.7.7	
				6.4 Draft Rev. 3 April 1996	GDC-19	Notes 1, 2, 3, 28, 29*			
Radiological Consequences of a Design Basis Loss-Of-Coolant-Accident: Leakage from Main Steam Isolation Valves	BWR EPUs that do not utilize alternative source term	SPSB		15.6.5, App. D Draft Rev. 2 April 1996	10 CFR 100	Notes 9, 12, 27*	2.7.5		
				6.4 Draft Rev. 3 April 1996	GDC-19	Notes 1, 2, 3, 28, 29*			
Radiological Consequences of Fuel Handling Accidents	EPUs that do not utilize alternative source term	SPSB	SPLB	15.7.4 Draft Rev. 2 April 1996	10 CFR 100 GDC-61	Notes 4, 5, 18, 19, 20, 27*	2.7.6	2.7.8	
				6.4 Draft Rev. 3 April 1996	GDC-19	Notes 1, 2, 3, 28, 29*			
Radiological Consequences of Spent Fuel Cask Drop Accidents	EPUs that do not utilize alternative source term	SPSB	EMEB SPLB	15.7.5 Draft Rev. 3 April 1996	10 CFR 100 GDC-61	Notes, 5, 16, 17, 8, 18, 27*	2.7.7	2.7.9	
				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, Rev. 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% of the Part 100 guideline values, even with conservative assumptions. The value of 10% should be replaced with 25%.

11. In Section III, “REVIEW PROCEDURES,” the guidance in the fourth paragraph, which deals with passive failures, should not be used.

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

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

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

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

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

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

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

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

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

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

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

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% 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 7

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
IEHB = Equipment and Human Performance Branch
PWR = pressurized-water reactor
RORP = Operating Reactor Improvements Program
SPLB = Plant Systems Branch
SPSB = Probabalistic Safety Assessment Branch
SRP = Standard Review Plan
SRXB = Reactor Systems Branch

ATTACHMENT 1 TO MATRIX 7

Independent Calculations

Radiological Consequences Analyses

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

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

MATRIX 8

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
							BWR	PWR	
Radiation Sources	All EPUs	IEHB		12.2 Draft Rev. 3 April 1996	10 CFR 20		2.8.1	2.8.1	
Radiation Protection Design Features	All EPUs	IEHB		12.3-4 Draft Rev. 3 April 1996	10 CFR 20 GDC-19		2.8.1	2.8.1	
Operational Radiation Protection Program	All EPUs	IEHB		12.5 Draft Rev. 3 April 1996	10 CFR 20	Note 1*	2.8.1	2.8.1	

Notes:

1. Regulatory Guide 8.14 was withdrawn on February 9, 2001, and should not be used.

LIST OF ACRONYMS FOR MATRIX 8

BWR = boiling-water reactor
CFR = *Code of Federal Regulations*
EPUs = extended power uprates
GDC = General Design Criterion
IEHB = Equipment and Human Performance Branch
PWR = pressurized-water reactor
SRP = Standard Review Plan

ATTACHMENT 1 TO MATRIX 8

Independent Calculations

Health Physics

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

MATRIX 9

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
							BWR	PWR	
Reactor Operator Training	All EPUs	IEHB		13.2.1 Draft Rev. 2 Dec. 2002	Specific review questions are provided in the template safety evaluations.		2.9	2.9	
Training for Non-Licensed Plant Staff	All EPUs	IEHB		13.2.2 Draft Rev. 2 Dec. 2002	Specific review questions are provided in the template safety evaluations.		2.9	2.9	
Operating and Emergency Operating Procedures	All EPUs	IEHB	SPLB SRXB	13.5.2.1 Draft Rev. 1 Dec. 2002	Specific review questions are provided in the template safety evaluations.		2.9	2.9	
Human Factors Engineering	All EPUs	IEHB		18.0 Draft Rev. 1 Dec. 2002	Specific review questions are provided in the template safety evaluations.		2.9	2.9	

LIST OF ACRONYMS FOR MATRIX 9

BWR = boiling-water reactor
EPUs = extended power uprates
IEHB = Equipment and Human Performance Branch
PWR = pressurized-water reactor
SPLB = Plant Systems Branch
SRP = Standard Review Plan
SRXB = Reactor Systems Branch

ATTACHMENT 1 TO MATRIX 9

Independent Calculations

Human Performance

Perform an independent calculation of operator response time based on the criteria of ANSI/ANS-58.8 for the worst-case (shortest times available) operator actions. This independent calculation is to be used for screening purposes only. Should the calculation indicate a timing issue, request the licensee to prove, through simulation or other means, that operators can successfully perform the action.

MATRIX 10

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
							BWR	PWR	
Power Ascension and Testing	All EPUs	IEHB	EEIB EMCB EMEB SPLB SPSB SRXB	14.2.1 Draft Rev. 0 Dec. 2002	Entire Section		2.10	2.10	

LIST OF ACRONYMS FOR MATRIX 10

BWR = boiling-water reactor
EEIB = Electrical & Instrumentation & Controls Branch
EMCB = Materials and Chemical Engineering Branch
EMEB = Mechanical & Civil Engineering Branch
EPUs = extended power uprates
IEHB = Equipment and Human Performance Branch
PWR = pressurized-water reactor
SPLB = Plant Systems Branch
SPSB = Probabalistic Safety Assessment Branch
SRP = Standard Review Plan
SRXB = Reactor Systems Branch

ATTACHMENT 1 TO MATRIX 10

Independent Calculations

Power Ascension and Testing Plan

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

MATRIX 11

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
							BWR	PWR	
Risk Evaluation	All EPUs	SPSB				Note 1* RG 1.174 RIS 2001-02	2.11	2.11	

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

LIST OF ACRONYMS FOR MATRIX 11

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

ATTACHMENT 1 TO MATRIX 11

Independent Calculations

Risk Evaluation

Use the guidance in Attachment 2 to Matrix 11 of RS-001 for determining when to perform independent calculations.

ATTACHMENT 2 TO MATRIX 11

Supplemental Risk Evaluation Review Guidance

Risk Evaluation

1. INTRODUCTION

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

Though power uprates are not submitted as risk-informed license applications, it is recognized that there are potential risk increases associated with implementing a power uprate due to the increased heat loads at higher powers and the resulting reductions in the times available to perform specific accident response actions. In addition, there can be impacts on the equipment loads and the potential for an increase in the frequency of reactor scrams due to these increased loads and tighter operating margins. For small power uprates (i.e., those referred to as measurement uncertainty recapture power uprates and stretch power uprates), the risk increases are expected to be exceedingly small. However, notwithstanding any plant modifications that could reduce risks, some increase in risk is expected for larger power uprates. Depending on the type of plant-specific modifications required to implement the larger power uprates, these power uprates have the potential for significantly increasing plant risks, especially if they significantly impact initiating event frequencies, component reliabilities, system success criteria, and/or operator response times. Further, large power uprate requests are specifically identified in Appendix D to SRP Chapter 19 as an example of the type of situation that might create "special circumstances" since they could "involve changes for which the synergistic or cumulative effects could significantly impact risk." Therefore, the Probabilistic Safety Assessment Branch (SPSB) Safety Program Section formally reviews all license application submittals for power uprates greater than 5 percent of their original licensed thermal power (OLTP) level.

As of December 2002, the SPSB Safety Program Section staff had performed risk reviews of eight extended power uprate license applications involving twelve units. All, but one, of these applications were for boiling water reactors (BWRs) of various design vintages, including: five BWR-3/Mark-I units (Monticello, Dresden 2 and 3, and Quad Cities 1 and 2), five BWR-4/Mark-I units (Hatch 1 and 2, Duane Arnold, and Brunswick 1 and 2), and one BWR-6/Mark III unit

(Clinton). The one pressurized water reactor (PWR) extended power uprate license application was for a Combustion Engineering (CE) plant with a large dry containment (Arkansas Nuclear One - Unit 2). The extended power uprates have been as high as 20 percent of OLTP.

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

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

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

2.   SCOPE OF REVIEW

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

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

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

3. RISK INFORMATION NEEDED FOR REVIEW

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

3.1 Internal Events Risk Information

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

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

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

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

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

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

3.2 External Events Risk Information

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

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

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

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

3.3 Shutdown Operations Risk Information

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

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

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

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

4. REVIEW GUIDANCE

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

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

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

For example, in addressing the risk impacts for shutdown operations in the absence of a licensee's shutdown PRA, the review staff should refer to SECY 97-168, "Issuance for Public Comment of Proposed Rulemaking Package for Shutdown and Fuel Storage Pool Operation," in which the staff provides estimates of shutdown risk for various interpretations of the industry guidance. The risk estimates cited in SECY 97-168 were not meant to bound plant operations, but were intended to be examples of reasonable interpretations of industry guidance. Depending on the specific licensee's approach to managing shutdown risks, an estimate of the magnitude of the risk for shutdown operations can be determined using SECY 97-168. An example of this review approach is provided as Attachment 3 to Matrix 11 of RS-001.

As a further example, in addressing the risk impacts related to seismic events for situations in which the licensee has performed a seismic margins analysis instead of a seismic PRA, the review staff may need to perform a simplistic calculation to determine the magnitude of the seismic risk. An approximation method is provided in a paper by Robert P. Kennedy entitled *"Overview of Methods for Seismic PRA and Margin Analysis Including Recent Innovations*," (Reference 6) that uses the plant's high confidence of a low probability of failure (HCLPF) value that is determined by the licensee's seismic margins analysis and the site's seismic hazard curve that is based on NUREG-1488 (Reference 7) to derive an approximation of the magnitude of the risk associated with seismic events. An example of this calculation is provided as Attachment 4 to Matrix 11 of RS-001.

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

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

5. RISK REVIEW PROCESS AND DOCUMENTATION

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

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

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

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

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

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

6. REFERENCES

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

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

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

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

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

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

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

ATTACHMENT 3 TO MATRIX 11

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

Risk Evaluation

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

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

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

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

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

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

ATTACHMENT 4 TO MATRIX 11

Example Staff Review of Seismic Risk Using Simplified Calculations

Risk Evaluation

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

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

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

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

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

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

SECTION 3

DOCUMENTATION OF REVIEW

3.1 Documenting Reviews of Extended Power Uprate Applications

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

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

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

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

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

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

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

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

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

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

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

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

SECTION 3.2 of RS-001

TEMPLATE SAFETY EVALUATION

for

BOILING-WATER REACTOR EXTENDED POWER UPRATE

TABLE OF CONTENTS

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]** includes the following special features/unique designs:

[Insert any special features/unique designs]

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

1.3 Licensee’s Approach

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

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

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

1.4 Plant Modifications

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

[Provide a list of plant modifications.]

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

1.5 Method of NRC Staff Review

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

In areas where the licensee and its contractors used previously approved or widely accepted methods in performing analyses related to the EPU, the NRC staff reviewed relevant material to ensure that the licensee/contractor used the methods consistent with the limitations and restrictions placed on the methods. In addition, the NRC staff considered changes in plant operating conditions on the use of these methods to ensure that the methods are appropriate for use at the 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 Reactor Systems

SEE INSERT 6 FOR SECTION 3.2 OF RS-001

2.7 Source Terms and Radiological Consequences Analyses

SEE INSERT 7 FOR SECTION 3.2 OF RS-001

2.8 Health Physics

SEE INSERT 8 FOR SECTION 3.2 OF RS-001

2.9 Human Performance

SEE INSERT 9 FOR SECTION 3.2 OF RS-001

2.10 Power Ascension and Testing Plan

SEE INSERT 10 FOR SECTION 3.2 OF RS-001

2.11 Risk Evaluation

SEE INSERT 11 FOR SECTION 3.2 OF RS-001

3.0 FACILITY OPERATING LICENSE AND TECHNICAL SPECIFICATION CHANGES

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

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

4.0 REGULATORY COMMITMENTS

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

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

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

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

5.0 RECOMMENDED AREAS FOR INSPECTION

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

[Provide list of recommended areas for inspection.]

6.0 STATE CONSULTATION

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

7.0 ENVIRONMENTAL CONSIDERATION

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

8.0 CONCLUSION

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

9.0 REFERENCES

1. RS-001, "Review Standard for Extended Power Uprates," December 2002.

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

Attachment: List of Acronyms

Principal Contributors:

Date:

LIST OF ACRONYMS

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

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

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

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

INSERT 1

FOR

SECTION 3.2 - BWR TEMPLATE SAFETY EVALUATION

2.1. Materials and Chemical Engineering

2.1.1. Reactor Vessel Material Surveillance Program

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.1.2. Pressure-Temperature Limits and Upper-Shelf Energy

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.1.3. Reactor Internal and Core Support Materials

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.1.4. Reactor Coolant Pressure Boundary Materials

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.1.5. Protective Coating Systems (Paints) - Organic Materials

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.1.6. Flow-Accelerated Corrosion

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusions

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

2.1.7. Reactor Water Cleanup System

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

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

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

INSERT 2

FOR

SECTION 3.2 - BWR TEMPLATE SAFETY EVALUATION

2.2 Mechanical and Civil Engineering

2.2.1. Pipe Rupture Locations and Associated Dynamic Effects

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.2.2. Pressure-Retaining Components and Component Supports

Regulatory Evaluation

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

Technical Evaluation

Nuclear Steam Supply System Piping, Components, and Supports

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

Balance-of-Plant Piping, Components, and Supports

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

Reactor Vessel and Supports

[Insert technical evaluation for reactor vessel and supports.]

Control Rod Drive Mechanism

[Insert technical evaluation for control rod drive mechanism.]

Recirculation Pumps and Supports

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

Conclusion

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

2.2.3. Reactor Pressure Vessel Internals and Core Supports

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.2.4. Safety-Related Valves and Pumps

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.2.5. Seismic and Dynamic Qualification of Mechanical and Electrical Equipment

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

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

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

INSERT 3

FOR

SECTION 3.2 - BWR TEMPLATE SAFETY EVALUATION

2.3 Electrical Engineering

2.3.1. Environmental Qualification of Electrical Equipment

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.3.2. Offsite Power System

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.3.3. AC Onsite Power System

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.3.4. DC Onsite Power System

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.3.5. Station Blackout

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

[2.3.6. Additional Review Areas (Electrical Engineering)]

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

INSERT 4

FOR

SECTION 3.2 - BWR TEMPLATE SAFETY EVALUATION

2.4. Instrumentation and Controls

2.4.1. Reactor Protection, Safety Features Actuation, and Control Systems

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

[2.4.2. Additional Review Areas (Instrumentation and Controls)]

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

INSERT 5

FOR

SECTION 3.2 - BWR TEMPLATE SAFETY EVALUATION

2.5 Plant Systems

2.5.1. Internal Hazards

2.5.1.1. Flooding

2.5.1.1.1. Flood Protection

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.5.1.1.2. Equipment and Floor Drains

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.5.1.1.3. Circulating Water System

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.5.1.2. Missile Protection

2.5.1.2.1. Internally Generated Missiles

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.5.1.2.2. Turbine Generator

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.5.1.3. Pipe Failures

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.5.1.4. Fire Protection

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.5.2. Containment Review Considerations

2.5.2.1. Primary Containment Functional Design

Regulatory Evaluation

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

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.5.2.2. Subcompartment Analyses

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.5.2.3. Mass and Energy Release

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

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's mass and energy release assessment and concludes that the licensee has adequately addresses 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.

2.5.2.4. Combustible Gas Control in Containment

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment related to combustible gas and concludes that the plant will continue to have sufficient capabilities, consistent with the requirements in 10 CFR 50.44, 10 CFR 50.46, and GDCs 41, 42, and 43 to prevent high concentrations of combustible gases in local areas, monitor combustible gas concentrations, and reduce combustible gas concentrations in the containment following implementation of the proposed EPU; and GDC-5 with respect to the use of shared systems. Therefore, the NRC staff finds the proposed EPU acceptable with respect to combustible gas control in containment.

2.5.2.5. Containment Heat Removal

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.5.2.6. Secondary Containment Functional Design

<u>Regulatory Evaluation</u>

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

<u>Technical Evaluation</u>

[Insert technical evaluation.]

<u>Conclusion</u>

The NRC staff has reviewed the licensee's assessment related to the secondary containment temperature and pressure 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.

2.5.3. Habitability, Filtration, and Fission Product Control

2.5.3.1. Control Room Habitability System

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.5.3.2. Engineered Safety Feature Atmosphere Cleanup

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.5.3.3. Fission Product Control Systems and Structures

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.5.3.4. Main Condenser Evacuation System

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.5.3.5. Turbine Gland Sealing System

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.5.3.6. Main Steam Isolation Valve Leakage Control System

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's evaluation 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, and 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.

2.5.4. Ventilation Systems

2.5.4.1. Control Room Area Ventilation System

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.5.4.2. Spent Fuel Pool Area Ventilation System

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.5.4.3. Auxiliary and Radwaste Area and Turbine Areas Ventilation Systems

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.5.4.4. Engineered Safety Feature Ventilation System

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.5.5. Component Cooling and Decay Heat Removal

2.5.5.1. Spent Fuel Pool Cooling and Cleanup System

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.5.5.2. Station Service Water System

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.5.5.3. Reactor Auxiliary Cooling Water Systems

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.5.5.4. Ultimate Heat Sink

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.5.6. Balance-of-Plant Systems

2.5.6.1. Main Steam

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.5.6.2. Main Condenser

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.5.6.3. Turbine Bypass

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.5.6.4. Condensate and Feedwater

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.5.7. Waste Management Systems

2.5.7.1. Gaseous Waste Management Systems

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.5.7.2. Liquid Waste Management Systems

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.5.7.3. Solid Waste Management Systems

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.5.8. Additional Considerations

2.5.8.1. Emergency Diesel Engine Fuel Oil Storage and Transfer System

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.5.8.2. Light Load Handling System (Related to Refueling)

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

[2.5.9. 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. Reactor Systems

2.6.1. Fuel System Design

Regulatory Evaluation

The fuel system consists of arrays of fuel rods, burnable poison rods, spacer grids and springs, end plates, channel boxes, and reactivity control rods. The NRC staff reviews the fuel system to ensure that (1) the fuel system is not damaged as a result of normal operation and anticipated operational occurrences, (2) fuel system damage is never so severe as to prevent control rod insertion when it is required, (3) the number of fuel rod failures is not underestimated for postulated accidents, and (4) coolability is always maintained. The NRC staff's review covers fuel system damage mechanisms, limiting values for important parameters, and performance of the fuel system during normal operation, anticipated operational occurrences, and postulated accidents. The NRC's acceptance criteria are based on (1) 10 CFR 50.46 for core cooling; (2) GDC-10 for assuring that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including anticipated operational occurrences; (3) GDC-27 for the reactivity control system being designed with appropriate margin, and in conjunction with the ECCS, being capable of controlling reactivity and cooling the core under postaccident conditions; and (4) GDC-35 for providing an ECCS to transfer heat from the reactor core following any loss of reactor coolant. Specific review criteria are contained in SRP Section 4.2 and other guidance provided in Matrix 6 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.6.2. Nuclear Design

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's analyses related to the effect of the proposed EPU on the nuclear design. 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.

2.6.3. Thermal and Hydraulic Design

Regulatory Evaluation

The NRC staff reviews the thermal and hydraulic design of the core and the RCS to confirm that the design (1) has been accomplished using acceptable analytical methods, (2) is equivalent to or a justified extrapolation from proven designs, (3) provides acceptable margins of safety from conditions which would lead to fuel damage during normal reactor operation and anticipated operational occurrences, and (4) is not susceptible to thermal-hydraulic instability. The review also covers hydraulic loads on the core and RCS components during normal operation and design-basis accident conditions and core thermal-hydraulic stability under normal operation and anticipated transients without scram (ATWS) events. The NRC's acceptance criteria are based on (1) GDC-10 for assuring that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including anticipated operational occurrences; and (2) GDC-12 for the reactor core and associated coolant, control, and protection systems being designed to assure that power oscillations, which can result in conditions exceeding specified acceptable fuel design limits, are not possible or can reliably and readily be detected and suppressed. Specific review criteria are contained in SRP Section 4.4 and other guidance provided in Matrix 6 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.6.4. Emergency Systems

2.6.4.1. Functional Design of Control Rod Drive System

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.6.4.2. Overpressure Protection During Power Operation

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.6.4.3. Reactor Core Isolation Cooling System

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.6.4.4. Residual Heat Removal System

Regulatory Evaluation

The RHR system is used to cool down the RCS following shutdown. The RHR system is typically a low pressure system which takes over the shutdown cooling function when the RCS temperature is reduced. The NRC staff's review covers the effect of the proposed EPU on the functional capability of the RHR system to cool the RCS following shutdown and provide decay heat removal. The NRC's acceptance criteria are based on (1) GDC-4 for dynamic effects associated with flow instabilities and loads (e.g., water hammer); (2) GDC-5 which requires that any sharing among nuclear power units of structures, systems and components important to safety will not significantly impair their safety function; (3) GDC-19 for control room requirements for normal operations and shutdown; and (4) GDC-34 which specifies requirements for an RHR system. Specific review criteria are contained in SRP Section 5.4.7 and other guidance provided in Matrix 6 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.6.4.5. Standby Liquid Control System

Regulatory Evaluation

The standby liquid control system (SLCS) provides backup capability for reactivity control independent of the control rod system. The SLCS functions by injecting a boron solution into the reactor to effect shutdown. The NRC staff's review covers the effect of the proposed EPU on the functional capability of the system to deliver the required amount of boron solution into the reactor. The NRC's acceptance criteria are based on (1) GDC-26 for the requirement that two independent reactivity control systems of different design principles be provided, and the requirement that one of the systems shall be capable of holding the reactor subcritical in the cold condition; (2) GDC-27 for the requirement that the reactivity control systems have a combined capability, in conjunction with poison addition by the ECCS, to reliably control reactivity changes under postulated accident conditions; and (3) 10 CFR 50.62(c)(4) for the SLCS being capable of reliably injecting a borated water solution into the reactor pressure vessel at a boron concentration, boron enrichment, and flow rate that provides sufficient reactivity control and for the system having automatic initiation, where required under the rule, to satisfy ATWS risk-reduction requirements. Specific review criteria are contained in SRP Section 9.3.5 and other guidance provided in Matrix 6 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.6.5. Accident and Transient Analyses

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

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.6.5.2. Decrease in Heat Removal by the Secondary System

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

Regulatory Evaluation

A number of initiating events may result in unplanned decreases in heat removal by the secondary system. These events result in a sudden reduction in steam flow and, consequently, result in pressurization events. Reactor protection and safety systems are actuated to mitigate the transient. The NRC staff's review covers the sequence of events, the analytical models used for analyses, the values of parameters used in the analytical models, and the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC-10 for the RCS being designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operations, including anticipated operational occurrences; (2) GDC-15 for the RCS and its associated auxiliaries being designed with appropriate margin to ensure that the RCPB will not be breached during normal operations, including anticipated operational occurrences; and (3) GDC-26 for the reliable control of reactivity changes to ensure that specified acceptable fuel design limits are not exceeded during normal operation, including anticipated operational occurrences. Specific review criteria are contained in SRP Section 15.2.1-5 and other guidance provided in Matrix 6 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.6.5.2.2. Loss of Nonemergency AC Power to the Station Auxiliaries

Regulatory Evaluation

The loss of nonemergency ac power is assumed to result in the loss of all power to the station auxiliaries and the simultaneous tripping of all reactor coolant circulation pumps. This causes a flow coastdown as well as a decrease in heat removal by the secondary system, a turbine trip, an increase in pressure and temperature of the coolant, and a reactor trip. Reactor protection and safety systems are actuated to mitigate the transient. The NRC staff's review covers (1) the sequence of events, (2) the analytical model used for analyses, (3) the values of parameters used in the analytical model, and (4) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC-10 for the RCS being designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operation including anticipated operational occurrences; (2) GDC-15 for the RCS and its associated auxiliaries being designed with appropriate margin to ensure that the RCPB will not be breached during normal operation including anticipated operational occurrences; and (3) GDC-26 for the reliable control of reactivity changes to assure ensure that specified acceptable fuel design limits are not exceeded during normal operation, including anticipated operational occurrences. Specific review criteria are contained in SRP Section 15.2.6 and other guidance provided in Matrix 6 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.6.5.2.3. Loss of Normal Feedwater Flow

Regulatory Evaluation

A loss of normal feedwater flow could occur from pump failures, valve malfunctions, or a LOOP. Loss of feedwater flow results in an increase in reactor coolant temperature and pressure which eventually requires a reactor trip to prevent fuel damage. Decay heat must be transferred from fuel following a loss of normal feedwater flow. Reactor protection and safety systems are actuated to provide this function and mitigate other aspects of the transient. The NRC staff's review covers (1) the sequence of events, (2) the analytical model used for analyses, (3) the values of parameters used in the analytical model, and (4) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC-10 for the RCS being designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operations including anticipated operational occurrences; (2) GDC-15 for the RCS and its associated auxiliaries being designed with appropriate margin to ensure that the RCPB will not be breached during normal operations including anticipated operational occurrences; and (3) GDC-26 for the reliable control of reactivity changes to ensure that specified acceptable fuel design limits are not exceeded during normal operation, including anticipated operational occurrences. Specific review criteria are contained in SRP Section 15.2.7 and other guidance provided in Matrix 6 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.6.5.3. Decrease in Reactor Coolant System Flow

2.6.5.3.1. Loss of Forced Reactor Coolant Flow

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.6.5.3.2. Reactor Recirculation Pump Rotor Seizure and Reactor Recirculation Pump Shaft Break

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.6.5.4. Reactivity and Power Distribution Anomalies

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

Regulatory Evaluation

An uncontrolled control rod assembly withdrawal from subcritical or low power startup condition may be caused by a malfunction of the reactor control or rod control systems. This withdrawal will uncontrollably add positive reactivity to the reactor core, resulting in a power excursion. The NRC staff's review covers (1) the description of the causes of the transient and the transient itself, (2) the initial conditions, (3) the reactor parameters used in the analysis, (4) the analytical methods and computer codes used, and (5) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC-10 for the RCS being designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operations, including anticipated operational occurrences; (2) GDC-20 for the reactor protection system being designed to initiate automatically the operation of appropriate systems, including the reactivity control systems, to ensure that specified acceptable fuel design limits are not exceeded during normal operation, including anticipated operational occurrences; and (3) GDC-25 for the functional design of redundant reactivity control systems to assure that specified acceptable fuel design limits are not exceeded in the event of a single malfunction of the reactivity control systems. Specific review criteria are contained in SRP Section 15.4.1 and other guidance provided in Matrix 6 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.6.5.4.2. Uncontrolled Control Rod Assembly Withdrawal at Power

Regulatory Evaluation

An uncontrolled control rod assembly withdrawal at power may be caused by a malfunction of the reactor control or rod control systems. This withdrawal will uncontrollably add positive reactivity to the reactor core, resulting in a power excursion. The NRC staff's review covers (1) the description of the causes of the anticipated operational occurrence and the description of the event itself, (2) the initial conditions, (3) the reactor parameters used in the analysis, (4) the analytical methods and computer codes used, and (5) the results of the associated analyses. The NRC's acceptance criteria are based on (1) GDC-10 for the RCS being designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operations, including anticipated operational occurrences; (2) GDC-20 for the reactor protection system being designed to initiate automatically the operation of appropriate systems, including the reactivity control systems, to ensure that specified acceptable fuel design limits are not exceeded during normal operation, including anticipated operational occurrences; and (3) GDC-25 for the functional design of redundant reactivity control systems to assure that specified acceptable fuel design limits are not exceeded in the event of a single malfunction of the reactivity control systems. Specific review criteria are contained in SRP Section 15.4.2 and other guidance provided in Matrix 6 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.6.5.4.3. 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 covers (1) the sequence of events, (2) the analytical model, (3) the values of parameters used in the analytical model, and (4) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC-10 and GDC-20 for the RCS being designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operations, including anticipated operational occurrences; (2) GDC-15 and GDC-28 for the RCS and its associated auxiliaries being designed with appropriate margin to ensure that the RCPB will not be breached during normal operations, including anticipated operational occurrences; and (3) GDC-26 for the reliable control of reactivity changes to ensure that specified acceptable fuel design limits are not exceeded during normal operations, including anticipated operational occurrences. Specific review criteria are contained in SRP Section 15.4.4-5 and other guidance provided in Matrix 6 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

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 specified acceptable fuel design limits and the RCPB pressure limits will not be exceeded as a result of this event. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 15, 20, 26, and 28 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the increase in core flow event.

2.6.5.4.4. Spectrum of Rod Drop Accidents

Regulatory Evaluation

The NRC staff evaluates the consequences of a control rod drop accident in the area of physics. The NRC staff's review covers the occurrences that lead to the accident, safety features designed to limit the amount of reactivity available and the rate at which reactivity can be added to the core, the analytical model used for analyses, and the results of the analyses. The NRC's acceptance criteria are based on GDC-28 for the effects of postulated reactivity accidents, neither resulting in damage to the RCPB greater than limited local yielding nor causing sufficient damage to impair significantly the capacity to cool the core. Specific review criteria are contained in SRP Section 15.4.9 and other guidance provided in Matrix 6 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.6.5.5. Inadvertent Operation of ECCS or Malfunction that Increases Reactor Coolant Inventory

Regulatory Evaluation

Equipment malfunctions, operator errors, and abnormal occurrences could cause unplanned increases in reactor coolant inventory. Depending on the temperature of the injected water and the response of the automatic control systems, a power level increase may result and, without adequate controls, could lead to fuel damage or overpressurization of the RCS. Alternatively, a power level decrease or depressurization may result. Reactor protection and safety systems are actuated to mitigate these events. The NRC staff's review covers (1) the sequence of events, (2) the analytical model used for analyses, (3) the values of parameters used in the analytical model, and (4) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC-10 for the RCS being designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operations, including anticipated operational occurrences; (2) GDC-15 for the RCS and its associated auxiliaries being designed with appropriate margin to ensure that the RCPB will not be breached during normal operations, including anticipated operational occurrences; and (3) GDC-26 for the reliable control of reactivity changes to ensure that specified acceptable fuel design limits are not exceeded during normal operation, including anticipated operational occurrences. Specific review criteria are contained in SRP Section 15.5.1-2 and other guidance provided in Matrix 6 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.6.5.6. Decrease in Reactor Coolant Inventory

2.6.5.6.1. Inadvertent Opening of a Pressure Relief Valve

Regulatory Evaluation

The inadvertent opening of a pressure relief valve results in a reactor coolant inventory decrease and a decrease in RCS pressure. The pressure relief valve discharges into the suppression pool. Normally there is no reactor trip. The pressure regulator senses the RCS pressure decrease and partially closes the turbine control valves (TCVs) to stabilize the reactor at a lower pressure. The reactor power settles out at nearly the initial power level. The coolant inventory is maintained by the feedwater control system using water from the condensate storage tank via the condenser hotwell. The NRC staff's review covers (1) the sequence of events, (2) the analytical model used for analyses, (3) the values of parameters used in the analytical model, and (4) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC-10 for the RCS being designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operations, including anticipated operational occurrences; (2) GDC-15 for the RCS and its associated auxiliaries being designed with appropriate margin to ensure that the RCPB will not be breached during normal operations, including anticipated operational occurrences; and (3) GDC-26 for the reactivity control systems to provide adequate control of reactivity changes to ensure that the acceptable fuel design limits are not exceeded during normal operations and anticipated transients during normal operations, inlcuding anticipated operational occurrences. Specific review criteria are contained in SRP Section 15.6.1 and other guidance provided in Matrix 6 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.6.5.6.2. Emergency Core Cooling System and Loss-of-Coolant Accidents

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.6.5.7. Anticipated Transients Without Scrams

Regulatory Evaluation

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

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

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

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

The NRC staff review is conducted to ensure that (1) the above requirements are met, (2) sufficient margin is available in the setpoint for the SLCS pump discharge relief valve such that SLCS operability is not affected, and (3) operator actions specified in the plant's Emergency Operating Procedures are consistent with the generic emergency procedure guidelines/severe accident guidelines (EPGs/SAGs). In addition, the NRC staff reviews the licensee's ATWS analysis to ensure that (1) the peak vessel bottom pressure is less than ASME Service Level C limit of 1500 psig; (2) the peak clad temperature is within the 10 CFR 50.46 limit of 2200 °F; (3) the peak suppression pool temperature is less than the design limit; and (4) the peak containment pressure is less than the containment design pressure. The NRC staff also evaluates the potential for thermal-hydraulic instability in conjunction with ATWS events using the methods and criteria approved by the NRC staff. For this analysis, the NRC staff reviews the limiting event determination, the sequence of events, the analytical model and its applicability, the values of parameters used in the analytical model, and the results of the analyses. If the licensee relies upon generic vendor analyses, the NRC staff reviews the licensee's justification of the applicability of that analysis to its plant and the operating conditions for the proposed EPU. Review guidance is provided in Matrix 6 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.6.6. <u>Fuel Storage</u>

2.6.6.1. New Fuel Storage

<u>Regulatory Evaluation</u>

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

<u>Technical Evaluation</u>

[Insert technical evaluation.]

<u>Conclusion</u>

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.

2.6.6.2. Spent Fuel Storage

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

[2.6.7. Additional Review Areas (Reactor Systems)]

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

INSERT 7

FOR

SECTION 3.2 - BWR TEMPLATE SAFETY EVALUATION

2.7 Source Terms and Radiological Consequences Analyses

2.7.1. Source Terms for Radwaste Systems Analyses

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

NOTE: Use Sections 2.7.2 and 2.7.3 below if the licensee's radiological consequences analyses are based on an alternative source term.

2.7.2. <u>Radiological Consequences Analyses Using Alternative Source Terms</u>

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.

<u>Regulatory Evaluation</u>

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

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

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

<u>Technical Evaluation</u>

[Insert technical evaluation.]

Conclusion

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

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

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

[2.7.3. Additional Review Areas (Radiological Consequences Analyses)]

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

NOTE: Use Sections 2.7.2 - 2.7.8 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)

2.7.2. Radiological Consequences of Control Rod Drop Accident

Regulatory Evaluation

The NRC staff reviews the analyses of the radiological consequences of a control rod drop accident (CRDA). The NRC staff's review includes an examination of (1) the plant's response to the accident, (2) the release of fission products from the core to the environment via the turbine and condensers as a result of the accident, (3) and the calculation of radiological doses at the exclusion area boundary (EAB) and low population zone (LPZ) outer boundary, and in the control room due to the releases from the accident. The NRC's acceptance criteria for the radiological consequences of a control rod drop accident are based on (1) GDC-19 for control room habitability and (2) 10 CFR Part 100 for mitigating the radiological consequences of an accident. Specific review criteria are contained in SRP Sections 6.4 and 15.4.9.A, and other guidance provided in Matrix 7 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

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

Regulatory Evaluation

The NRC staff reviews the analysis of the radiological consequences of failures outside the containment of small lines connected to the primary coolant pressure boundary (e.g., instrument lines and sample lines). The NRC staff's review includes (1) the identification of small lines postulated to fail and the isolation provisions for these lines; (2) the failure scenario; (3) the models and assumptions for the calculation of the radiological doses for the postulated failure; and (4) an evaluation of the primary coolant iodine activity, including the effects of a concurrent iodine spike, and the TSs for the reactor coolant iodine activity. The NRC's acceptance criteria for the radiological consequences of failures outside the containment of small lines connected to the primary coolant pressure boundary are based on (1) GDC-19 for control room habitability and (2) GDC-55 for the isolation requirements of small-diameter lines connected to the primary system that are acceptable on the basis of meeting 10 CFR 100.11. Specific review criteria are contained in SRP Sections 6.4 and 15.6.2, and other guidance provided in Matrix 7 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.7.4. Radiological Consequences of Main Steamline Failure Outside Containment

Regulatory Evaluation

The NRC staff reviews the analyses of the radiological consequences of an MSLB accident outside the containment to ensure that radioactive releases due to the failure are adequately limited by the TS limit on primary coolant activity. The NRC staff's review includes two cases for the reactor coolant iodine concentration: (1) with a preaccident iodine spike and (2) with the maximum equilibrium concentration for continued full-power operation. The NRC's acceptance criteria for the radiological consequences of an MSLB outside containment are based on (1) GDC-19 for control room habitability and (2) 10 CFR Part 100 for the radiological consequences of an accident. Specific review criteria are contained in SRP Sections 6.4 and 15.6.4, and other guidance provided in Matrix 7 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.7.5. Radiological Consequences of a Design-Basis Loss-of-Coolant Accident

Regulatory Evaluation

The NRC staff reviews the analyses of the radiological consequences of a design-basis LOCA. This review includes a summary review of the doses from the hypothetical design-basis LOCA and a specific review of the doses from containment leakage and leakage from ESF components outside containment that contribute to the total LOCA doses. The NRC staff's review also includes (1) the contribution to the dose due to leakage from the main steam isolation valves (MSIVs); (2) the methodology and results of calculations of the radiological consequences resulting from containment and ESF components and MSIV leakage following a hypothetical LOCA; and (3) an assessment of the containment with respect to the assumptions and the input parameters for the dose calculations. The NRC's calculations are based on pertinent information in the safety analysis report (SAR) and considers the NRC staff's evaluation of dose-mitigating ESFs. The NRC's acceptance criteria for the radiological consequences of a design-basis LOCA are based on (1) GDC-19 for control room habitability and (2) 10 CFR Part 100 for mitigating the radiological consequences of an accident. Specific review criteria are contained in SRP Section 6.4 and Appendices A, B, and D of SRP Section 15.6.5, and other guidance provided in Matrix 7 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.7.6. Radiological Consequences of Fuel Handling Accidents

Regulatory Evaluation

The NRC staff reviews the analyses of the radiological consequences of a postulated FHA. The purpose of this review is to evaluate the adequacy of system design features and plant procedures provided for the mitigation of the radiological consequences of accidents that involve damage to spent fuel. Such accidents include the dropping of a single fuel assembly and handling tool or a heavy object onto other spent fuel assemblies. Such accidents may occur inside the containment, along the fuel transfer canal, and in the fuel building. The NRC staff's review includes (1) the sequence of events, models, and assumptions used by the licensee for the calculation of the radiological doses; (2) the adequacy of the ESFs provided for the purpose of mitigating potential accident doses; and (3) the containment ventilation system with respect to its function as a dose-mitigating ESF system, including the radiation detection system on the containment purge/vent lines for those plants that will vent or purge the containment during fuel handling operations. The NRC's acceptance criteria for the radiological consequences of FHAs are based on (1) GDC-19 for control room habitability; (2) GDC-61 for appropriate containment, confinement, and filtering systems; and (3) 10 CFR Part 100 for the calculated radiological consequences of FHAs. Specific review criteria are contained in SRP Sections 6.4 and 15.7.4, and other guidance provided in Matrix 7 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.7.7. Radiological Consequences of Spent Fuel Cask Drop Accidents

Regulatory Evaluation

The NRC staff reviews the analyses of the radiological consequences of the release of fission products from irradiated fuel in a spent fuel cask that is postulated to drop during cask handling operations. The NRC staff's review is conducted to verify various design and operation aspects of the system. The NRC staff's review includes (1) determining a need for a design-basis radiological analysis sequence of events; (2) models and assumptions used by the licensee for the calculation of the radiological doses; (3) comparing calculated doses to exposure guidelines to determine the acceptability of the EAB and LPZ outer boundary distances and to confirm the adequacy of ESFs provided for the purpose of mitigating potential doses from spent fuel cask drop accidents, including the effects on control room habitability; and (4) examining the relationship of the operational modes of the standby gas treatment system (SGTS) to the time sequence of the accident in order to give proper credit, in a dual containment design where the fuel building may be exhausted through the SGTS. The NRC's acceptance criteria for the radiological consequences of spent fuel cask drop accidents are based on (1) GDC-19 for control room habitability; (2) GDC-61 for appropriate containment, confinement and filtering systems; and (3) 10 CFR Part 100 for the calculated radiological consequences of a spent fuel cask drop accident. Specific review criteria are contained in SRP Sections 6.4 and 15.7.5, and other guidance provided in Matrix 7 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

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

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

INSERT 8

FOR

SECTION 3.2 - BWR TEMPLATE SAFETY EVALUATION

2.8 Health Physics

2.8.1. Occupational and Public Radiation Doses

Regulatory Evaluation

The NRC staff conducts its review in this area to ascertain what overall effects the proposed EPU will have on both occupational and public radiation doses and to determine that the licensee has taken the necessary steps to ensure that any dose increases will be maintained as low as is reasonably achievable. The NRC staff's review includes an evaluation of any increases in radiation sources and how this may affect plant area dose rates, plant radiation zones, and plant area accessibility. The NRC staff evaluates how personnel doses needed to access plant vital areas following an accident are affected. The NRC staff considers the effects of the proposed EPU on nitrogen-16 levels in the plant and any effects this increase may have on radiation doses outside the plant and at the site boundary from skyshine. The NRC'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 8 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

[2.8.2. Additional Review Areas (Health Physics)]

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

INSERT 9

FOR

SECTION 3.2 - BWR TEMPLATE SAFETY EVALUATION

2.9 Human Performance

2.9.1. Human Factors

Regulatory Evaluation

The area of human factors deals with programs, procedures, training, and plant design features related to operator performance during normal and accident conditions. The NRC staff's human factors evaluation is conducted to ensure that operator performance is not adversely affected as a result of system changes required for the proposed EPU. The NRC staff's review covers changes to operator actions, human-system interfaces, and procedures and training required for the proposed EPU. The NRC's acceptance criteria for human factors are based on GDC-19, 10 CFR 50.54(i) and (m), 10 CFR 50.59, 10 CFR 50.120, and 10 CFR 55.59. Specific review criteria are contained in SRP Sections 13.2.1, 13.2.2, 13.5.2.1, and 18.0.

Technical Evaluation

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

1. Changes in Emergency and Abnormal Operating Procedures

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

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

2. Changes to Operator Actions Sensitive to Power Uprate

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

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

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

3. Changes to Control Room Controls, Displays and Alarms

   Describe any changes the proposed EPU will have on the operator interfaces for control room controls, displays and alarms. For example, what zone markings (e.g. normal, marginal and out-of-tolerance ranges) on meters will change? What setpoints will change? How will the operators know of the change? Describe any controls, displays, alarms that will be upgraded from analog to digital instruments as a result of the proposed EPU and how operators were tested to determine they could use the instruments reliably. (SRP Section 18.0)

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

4. Changes on the Safety Parameter Display System

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

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

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

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

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

Conclusion

The NRC staff has reviewed the changes to operator actions, human-system interfaces, procedures and training required for the proposed EPU and concludes that (1) the licensee has appropriately accounted for the effects of the proposed EPU on the available time for operator actions and (2) the licensee has taken appropriate actions to ensure that operators' performance is not adversely affected by the proposed EPU. The NRC staff further concludes that the licensee will continue to meet the requirements of 10 CFR 50.54(i) and (m), 10 CFR 50.59, 10 CFR 50.120, and 10 CFR 55.59 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.

[2.9.2. Additional Review Areas (Human Performance)]

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

INSERT 10

FOR

SECTION 3.2 - BWR TEMPLATE SAFETY EVALUATION

2.10 Power Ascension and Testing Plan

2.10.1. Approach to EPU Power Level and Test Plan

Regulatory Evaluation

The purpose of the power ascension and testing plan is to demonstrate that modifications to the plant are adequately designed and implemented and that the plant can be operated safely at the proposed EPU power level. The test program also provides additional assurance that the requested power uprate does not invalidate principle design criteria contained in the original licensing basis. The NRC staff's review includes an evaluation of: (1) plans for the initial approach to the proposed maximum licensed thermal power level, including verification of adequate plant performance, (2) transient testing requirements necessary to demonstrate that the plant can be operated safely 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 power ascension and testing plan are based on 10 CFR Part 50, Appendix B, Criterion XI, for the performance of all testing required to demonstrate that SSCs will perform satisfactorily in service. Specific review criteria are contained in SRP Section 14.2.1.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the proposed EPU test plan related to initial approach to the proposed power level, steady state-performance, and transient testing, and concludes that the proposed plan will demonstrate that modifications made to the plant have been adequately designed and implemented and that the plant can be safely operated at the proposed power level. The NRC staff further concludes that the proposed EPU testing plan satisfies the requirements of 10 CFR Part 50, Appendix B, Criterion XI. Therefore, the NRC staff finds the proposed EPU test plan acceptable.

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

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

INSERT 11

FOR

SECTION 3.2 - BWR TEMPLATE SAFETY EVALUATION

2.11. Risk Evaluation

2.11.1 Risk Evaluation of Extended Power Uprate

Regulatory Evaluation

A risk evaluation is conducted to (1) demonstrate that the risks associated with the proposed EPU are acceptable and (2) determine if “special circumstances” are created by the proposed EPU. As described in Appendix D of SRP Chapter 19, special circumstances are any issues that would potentially rebut the presumption of adequate protection provided by the licensee to meet the deterministic requirements and regulations. The NRC staff’s review covers the impact of the proposed EPU on core damage frequency (CDF) and large early release frequency (LERF) for the plant due to changes in the risks associated with internal events, external events, and shutdown operations. In addition, the NRC staff’s review covers the quality of the risk analyses used by the licensee to support the application for the proposed EPU. This includes a review of licensee actions to address issues or weaknesses that may have been raised in previous NRC staff reviews of the licensee’s individual plant examinations (IPEs) and individual plant examinations of external events (IPEEE), or by an industry peer review. The NRC’s risk acceptability guidelines are contained in RG 1.174. Specific review guidance is contained in Matrix 11 of RS-001 and its attachments.

Technical Evaluation

[Insert technical evaluation]

Conclusion

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

[2.11.2. 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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<u>SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION</u>

<u>RELATED TO AMENDMENT NO. TO FACILITY OPERATING LICENSE NO. **[XXX-XX]**</u>

<u>**[NAME OF LICENSEE]**</u>

<u>**[NAME OF FACILITY]**</u>

<u>DOCKET NO. 50-**[XXX]**</u>

1.0 <u>INTRODUCTION</u>

1.1 <u>Application</u>

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 <u>Background</u>

[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]** includes the following special features/unique designs:

[Insert any special features/unique designs]

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

1.3 Licensee's Approach

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

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

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

1.4 Plant Modifications

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

[Provide a list of plant modifications.]

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

1.5 Method of NRC Staff Review

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

In areas where the licensee and its contractors used previously approved or widely accepted methods in performing analyses related to the EPU, the NRC staff reviewed relevant material to ensure that the licensee/contractor used the methods consistent with the limitations and restrictions placed on the methods. In addition, the NRC staff considered changes in plant operating conditions on the use of these methods to ensure that the methods are appropriate for use at the 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

INSERT 5 FOR SECTION 3.3 OF RS-001

2.6 Reactor Systems

INSERT 6 FOR SECTION 3.3 OF RS-001

2.7 Source Terms and Radiological Consequences Analyses

INSERT 7 FOR SECTION 3.3 OF RS-001

2.8 Health Physics

INSERT 8 FOR SECTION 3.3 OF RS-001

2.9 Human Performance

INSERT 9 FOR SECTION 3.3 OF RS-001

2.10 Power Ascension and Testing Plan

INSERT 10 FOR SECTION 3.3 OF RS-001

2.11 Risk Evaluation

INSERT 11 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 <u>REGULATORY COMMITMENTS</u>

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 <u>RECOMMENDED AREAS FOR INSPECTION</u>

The NRC staff has conducted an extensive review of the licensee's plans and analyses related to the proposed EPU and concluded that they are acceptable. The NRC review staff has identified the following areas for consideration by the NRC inspection staff during the licensee's implementation of the proposed EPU. These areas are recommended based on past experience with EPUs, the extent and unique nature of modifications required to achieve the EPU, and new conditions of operation required for the 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 <u>STATE CONSULTATION</u>

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 <u>ENVIRONMENTAL CONSIDERATION</u>

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

8.0 CONCLUSION

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

9.0 REFERENCES

1. RS-001, "Review Standard for Extended Power Uprates," December 2002.

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

Attachment: List of Acronyms

Principal Contributors:

Date:

LIST OF ACRONYMS

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

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

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

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

INSERT 1

FOR

SECTION 3.3 - PWR TEMPLATE SAFETY EVALUATION

2.1. Materials and Chemical Engineering

2.1.1. Reactor Vessel Material Surveillance Program

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.1.2. Pressure-Temperature Limits and Upper-Shelf Energy

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.1.3. Pressurized Thermal Shock

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.1.4. Reactor Internal and Core Support Materials

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.1.5. Reactor Coolant Pressure Boundary Materials

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.1.6. Leak-Before-Break

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.1.7. Protective Coating Systems (Paints) - Organic Materials

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.1.8. Flow-Accelerated Corrosion

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusions

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

2.1.9. Steam Generator Tube Inservice Inspection

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.1.10. Steam Generator Blowdown System

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.1.11. Chemical and Volume Control System

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

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

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

INSERT 2

FOR

SECTION 3.3 - PWR TEMPLATE SAFETY EVALUATION

2.2 Mechanical and Civil Engineering

2.2.1. Pipe Rupture Locations and Associated Dynamic Effects

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.2.2. Pressure-Retaining Components and Component Supports

Regulatory Evaluation

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

Technical Evaluation

Nuclear Steam Supply System Piping, Components, and Supports

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

Balance-of-Plant Piping, Components, and Supports

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

Reactor Vessel and Supports

[Insert technical evaluation for reactor vessel and supports.]

Control Rod Drive Mechanism

[Insert technical evaluation for control rod drive mechanism.]

Steam Generators and Supports

[Insert technical evaluation for steam generators and supports.]

Reactor Coolant Pumps and Supports

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

Pressurizer and Supports

[Insert technical evaluation for pressurizer and supports.]

Conclusion

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

2.2.3. Reactor Pressure Vessel Internals and Core Supports

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.2.4. Safety-Related Valves and Pumps

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.2.5. Seismic and Dynamic Qualification of Mechanical and Electrical Equipment

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

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

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

INSERT 3

FOR

SECTION 3.3 - PWR TEMPLATE SAFETY EVALUATION

2.3 Electrical Engineering

2.3.1. Environmental Qualification of Electrical Equipment

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.3.2. Offsite Power System

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.3.3. AC Onsite Power System

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.3.4. DC Onsite Power System

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.3.5. Station Blackout

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

[2.3.6. Additional Review Areas (Electrical Engineering)]

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

INSERT 4

FOR

SECTION 3.3 - PWR TEMPLATE SAFETY EVALUATION

2.4. Instrumentation and Controls

2.4.1. Reactor Protection, Safety Features Actuation, and Control Systems

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

[2.4.2. Additional Review Areas (Instrumentation and Controls)]

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

INSERT 5

FOR

SECTION 3.3 - PWR TEMPLATE SAFETY EVALUATION

2.5 Plant Systems

2.5.1. Internal Hazards

2.5.1.1. Flooding

2.5.1.1.1. Flood Protection

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.5.1.1.2. Equipment and Floor Drains

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.5.1.1.3. Circulating Water System

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.5.1.2. Missile Protection

2.5.1.2.1. Internally Generated Missiles

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.5.1.2.2. Turbine Generator

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.5.1.3. Pipe Failures

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.5.1.4. Fire Protection

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.5.2. Containment Review Considerations

2.5.2.1. Primary Containment Functional Design

Regulatory Evaluation

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

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

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

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

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

<u>Technical Evaluation</u>

[Insert technical evaluation.]

<u>Conclusion</u>

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 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.

2.5.2.2. Subcompartment Analyses

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.5.2.3. Mass and Energy Release

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

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.5.2.3.2. Mass and Energy Release Analysis for Secondary System Pipe Ruptures

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.5.2.4. Combustible Gas Control in Containment

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's assessment related to combustible gas and concludes that the plant will continue to have sufficient capabilities, consistent with the requirements in 10 CFR 50.44,10 CFR 50.46, and GDCs 41, 42, and 43 to prevent high concentrations of combustible gases in local areas, monitor combustible gas concentrations, and reduce combustible gas concentrations in the containment following implementation of the proposed EPU; and GDC-5 with respect to the use of shared systems. Therefore, the NRC staff finds the proposed EPU acceptable with respect to combustible gas control in containment.

2.5.2.5. Containment Heat Removal

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.5.2.6. Pressure Analysis for ECCS Performance Capability

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.5.2.7. Pressurizer Relief Tank

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.5.3. Habitability, Filtration, and Fission Product Control

2.5.3.1. Control Room Habitability System

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.5.3.2. ESF Atmosphere Cleanup

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.5.3.3. Fission Product Control Systems and Structures

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.5.3.4. Main Condenser Evacuation System

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.5.3.5. Turbine Gland Sealing System

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.5.4. Ventilation Systems

2.5.4.1. Control Room Area Ventilation System

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.5.4.2. Spent Fuel Pool Area Ventilation System

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.5.4.3. Auxiliary and Radwaste Area and Turbine Area Ventilation Systems

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.5.4.4. Engineered Safety Feature (ESF) Ventilation System

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.5.5. Component Cooling and Decay Heat Removal

2.5.5.1. Spent Fuel Pool Cooling and Cleanup System

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.5.5.2. Station Service Water System

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.5.5.3. Reactor Auxiliary Cooling Water Systems

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.5.5.4. Ultimate Heat Sink

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.5.5.5. Auxiliary Feedwater System

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.5.6. Balance-of-Plant Systems

2.5.6.1. Main Steam

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.5.6.2. Main Condenser

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.5.6.3. Turbine Bypass

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.5.6.4. Condensate and Feedwater

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.5.7. Waste Management Systems

2.5.7.1. Gaseous Waste Management Systems

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.5.7.2. Liquid Waste Management Systems

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.5.7.3. Solid Waste Management Systems

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.5.8. Additional Considerations

2.5.8.1. Emergency Diesel Engine Fuel Oil Storage and Transfer System

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.5.8.2. Light Load Handling System (Related to Refueling)

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

[2.5.9. 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. Reactor Systems

2.6.1. Fuel System Design

Regulatory Evaluation

The fuel system consists of arrays of fuel rods, burnable poison rods, spacer grids and springs, end plates, channel boxes, and reactivity control rods. The NRC staff reviews the fuel system to ensure that (1) the fuel system is not damaged as a result of normal operation and anticipated operational occurrences, (2) fuel system damage is never so severe as to prevent control rod insertion when it is required, (3) the number of fuel rod failures is not underestimated for postulated accidents, and (4) coolability is always maintained. The NRC staff's review covers fuel system damage mechanisms, limiting values for important parameters, and performance of the fuel system during normal operation, anticipated operational occurrences, and postulated accidents. The NRC's acceptance criteria are based on (1) 10 CFR 50.46 for core cooling; (2) GDC-10 for assuring that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including anticipated operational occurrences; (3) GDC-27 for the reactivity control system being designed with appropriate margin, and in conjunction with the ECCS, being capable of controlling reactivity and cooling the core under postaccident conditions; and (4) GDC-35 for providing an ECCS to transfer heat from the reactor core following any loss of reactor coolant. Specific review criteria are contained in SRP Section 4.2 and other guidance provided in Matrix 6 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.6.2. Nuclear Design

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the licensee's analyses related to the effect of the proposed EPU on the nuclear design. 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.

2.6.3. Thermal and Hydraulic Design

Regulatory Evaluation

The NRC staff reviews the thermal and hydraulic design of the core and the RCS to confirm that the design (1) has been accomplished using acceptable analytical methods, (2) is equivalent to or a justified extrapolation from proven designs, (3) provides acceptable margins of safety from conditions which would lead to fuel damage during normal reactor operation and anticipated operational occurrences, and (4) is not susceptible to thermal-hydraulic instability. The review also covers hydraulic loads on the core and RCS components during normal operation and design-basis accident conditions and core thermal-hydraulic stability under conditions of normal operation and anticipated operational occurrences. The NRC's acceptance criteria are based on GDC-10 for the reactor core being designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during normal operation or anticipated operational occurrences. Specific review criteria are contained in SRP Section 4.4 and other guidance provided in Matrix 6 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.6.4. Emergency Systems

2.6.4.1. Functional Design of Control Rod Drive System

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.6.4.2. Overpressure Protection During Power Operation

Regulatory Evaluation

Overpressure protection for the RCPB during power operation is provided by relief and safety valves and the reactor protection system. The NRC staff's review covers pressurizer relief and safety valves and the piping from these valves to the quench tank and RCS relief and safety valves. The NRC's acceptance criteria are based on (1) GDC-15 for the RCS and associated auxiliary, control, and protection systems being designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including anticipated operational occurrences and (2) GDC-31 for the RCPB being designed with sufficient margin to assure that it behaves in a nonbrittle manner and that the probability of rapidly propagating fracture is minimized. Specific review criteria are contained in SRP Section 5.2.2 and other guidance provided in Matrix 6 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.6.4.3. Overpressure Protection During Low Temperature Operation

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.6.4.4. Residual Heat Removal System

Regulatory Evaluation

The residual heat removal (RHR) system is used to cool down the RCS following shutdown. The RHR system is typically a low pressure system which takes over the shutdown cooling function when the RCS temperature is reduced. The NRC staff's review covers the effect of the proposed EPU on the functional capability of the RHR system to cool the RCS following shutdown and provide decay heat removal. The NRC's acceptance criteria are based on (1) GDC-4 for dynamic effects associated with flow instabilities and loads (e.g., water hammer); (2) GDC-5 which requires that any sharing among nuclear power units of structures, systems and components important to safety will not significantly impair their safety function; (3) GDC-19 for control room requirements for normal operations and shutdown; and (4) GDC-34 which specifies requirements for an RHR system. Specific review criteria are contained in SRP Section 5.4.7 and other guidance provided in Matrix 6 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.6.5. Accident and Transient Analyses

2.6.5.1. Increase in Heat Removal by the Secondary System

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

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.6.5.1.2. Steam System Piping Failures Inside and Outside Containment

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.6.5.2. Decrease in Heat Removal By the Secondary System

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

Regulatory Evaluation

A number of initiating events may result in unplanned decreases in heat removal by the secondary system. These events result in a sudden reduction in steam flow and consequently result in pressurization events. Reactor protection and safety systems are actuated to mitigate the transient. The NRC staff's review covers the sequence of events, the analytical models used for analyses, the values of parameters used in the analytical models, and the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC-10 for the RCS being designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operations, including anticipated operational occurrences; (2) GDC-15 for the RCS and its associated auxiliaries being designed with appropriate margin to ensure that the RCPB will not be breached during normal operations, including anticipated operational occurrences; and (3) GDC-26 for the reliable control of reactivity changes to ensure that specified acceptable fuel design limits are not exceeded during normal operation, including anticipated operational occurrences. Specific review criteria are contained in SRP Section 15.2.1-5 and other guidance provided in Matrix 6 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.6.5.2.2. Loss of Nonemergency AC Power to the Station Auxiliaries

Regulatory Evaluation

The loss of nonemergency ac power is assumed to result in the loss of all power to the station auxiliaries and the simultaneous tripping of all reactor coolant circulation pumps. This causes a flow coastdown, as well as a decrease in heat removal by the secondary system, a turbine trip, an increase in pressure and temperature of the coolant, and a reactor trip. Reactor protection and safety systems are actuated to mitigate the transient. The NRC staff's review covers (1) the sequence of events, (2) the analytical model used for analyses, (3) the values of parameters used in the analytical model, and (4) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC-10 for the RCS being designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operation, including anticipated operational occurrences; (2) GDC-15 for the RCS and its associated auxiliaries being designed with appropriate margin to ensure that the RCPB will not be breached during normal operation, including anticipated operational occurrences; and (3) GDC-26 for the reliable control of reactivity changes to assure ensure that specified acceptable fuel design limits are not exceeded during normal operation, including anticipated operational occurrences. Specific review criteria are contained in SRP Section 15.2.6 and other guidance provided in Matrix 6 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.6.5.2.3. Loss of Normal Feedwater Flow

Regulatory Evaluation

A loss of normal feedwater flow could occur from pump failures, valve malfunctions, or a loss of offsite power (LOOP). Loss of feedwater flow results in an increase in reactor coolant temperature and pressure, which eventually requires a reactor trip to prevent fuel damage. Decay heat must be transferred from fuel following a loss of normal feedwater flow. Reactor protection and safety systems are actuated to provide this function and mitigate other aspects of the transient. The NRC staff's review covers (1) the sequence of events, (2) the analytical model used for analyses, (3) the values of parameters used in the analytical model, and (4) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC-10 for the RCS being designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operations, including anticipated operational occurrences; (2) GDC-15 for the RCS and its associated auxiliaries being designed with appropriate margin to ensure that the RCPB will not be breached during normal operations, including anticipated operational occurrences; and (3) GDC-26 for the reliable control of reactivity changes to ensure that specified acceptable fuel design limits are not exceeded during normal operation, including anticipated operational occurrences. Specific review criteria are contained in SRP Section 15.2.7 and other guidance provided in Matrix 6 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.6.5.2.4. Feedwater System Pipe Breaks Inside and Outside Containment

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.6.5.3. Decrease in Reactor Coolant System Flow

2.6.5.3.1. Loss of Forced Reactor Coolant Flow

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.6.5.3.2. Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.6.5.4. Reactivity and Power Distribution Anomalies

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

Regulatory Evaluation

An uncontrolled control rod assembly withdrawal from subcritical or low power startup condition may be caused by a malfunction of the reactor control or rod control systems. This withdrawal will uncontrollably add positive reactivity to the reactor core, resulting in a power excursion. The NRC staff's review covers (1) the description of the causes of the transient and the transient itself, (2) the initial conditions, (3) the reactor parameters used in the analysis, (4) the analytical methods and computer codes used, and (5) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC-10 for the RCS being designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operations, including anticipated operational occurrences; (2) GDC-20 for the reactor protection system being designed to initiate automatically the operation of appropriate systems, including the reactivity control systems, to ensure that specified acceptable fuel design limits are not exceeded during normal operation, including anticipated operational occurrences; and (3) GDC-25 for the functional design of redundant reactivity control systems to assure that specified acceptable fuel design limits are not exceeded in the event of a single malfunction of the reactivity control systems. Specific review criteria are contained in SRP Section 15.4.1 and other guidance provided in Matrix 6 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.6.5.4.2. Uncontrolled Control Rod Assembly Withdrawal at Power

Regulatory Evaluation

An uncontrolled control rod assembly withdrawal at power may be caused by a malfunction of the reactor control or rod control systems. This withdrawal will uncontrollably add positive reactivity to the reactor core, resulting in a power excursion. The NRC staff's review covers (1) the description of the causes of the anticipated operational occurrence and the description of the event itself, (2) the initial conditions, (3) the reactor parameters used in the analysis, (4) the analytical methods and computer codes used, and (5) the results of the associated analyses. The NRC's acceptance criteria are based on (1) GDC-10 for the RCS being designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operations, including anticipated operational occurrences; (2) GDC-20 as it relates the reactor protection system being designed to initiate automatically the operation of appropriate systems, including the reactivity control systems, to ensure that specified acceptable fuel design limits are not exceeded during normal operation, including anticipated operational occurrences; and (3) GDC-25 for the functional design of redundant reactivity systems to assure that specified acceptable fuel design limits are not exceeded in the event of a single malfunction of the reactivity control systems. Specific review criteria are contained in SRP Section 15.4.2 and other guidance provided in Matrix 6 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.6.5.4.3. Control Rod Misoperation

Regulatory Evaluation

The NRC staff's review covers the types of control rod misoperations that are assumed to occur, including those caused by a system malfunction or operator error. The review covers (1) descriptions of rod position, flux, pressure, and temperature indication systems, and those actions initiated by these systems (e.g., turbine runback, rod withdrawal prohibit, rod block) which can mitigate the effects or prevent the occurrence of various misoperations; (2) the sequence of events; (3) the analytical model used for analyses; (4) important inputs to the calculations; and (5) the results of the analyses. The NRC's acceptance criteria are based on (1) GDC-10 for the RCS being designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operations, including anticipated operational occurrences; (2) GDC-20 for the reactor protection system being designed to automatically initiate appropriate systems to ensure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences; and (3) GDC-25 for the functional design of redundant reactivity systems to assure that specified acceptable fuel design limits are not exceeded in the event of a single malfunction of the reactivity control systems. Specific review criteria are contained in SRP Section 15.4.3 and other guidance provided in Matrix 6 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.6.5.4.4. Startup of an Inactive Loop at an Incorrect Temperature

Regulatory Evaluation

A startup of an inactive loop transient may result in either an increased core flow or the introduction of cooler or deborated water into the core. This event causes an increase in core reactivity due to decreased moderator temperature or moderator boron concentration. The NRC staff's review covers (1) the sequence of events, (2) the analytical model, (3) the values of parameters used in the analytical model, and (4) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC-10 and GDC-20 for the RCS being designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operations, including anticipated operational occurrences; (2) GDC-15 and GDC-28 for the RCS and its associated auxiliaries being designed with appropriate margin to ensure that the RCPB will not be breached during normal operations, including anticipated operational occurrences; and (3) GDC-26 for the reliable control of reactivity changes to ensure that specified acceptable fuel design limits are not exceeded during normal operations, including anticipated operational occurrences. Specific review criteria are contained in SRP Section 15.4.4-5 and other guidance provided in Matrix 6 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

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

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.6.5.4.6. Spectrum of Rod Ejection Accidents

Regulatory Evaluation

Control rod ejection accidents cause a rapid positive reactivity insertion together with an adverse core power distribution, which could lead to localized fuel rod damage. The NRC staff evaluates the consequences of a control rod ejection accident to determine the potential damage caused to the RCPB and to determine whether the fuel damage resulting from such an accident could impair cooling water flow. The NRC staff's review covers initial conditions, rod patterns and worths, scram worth as a function of time, reactivity coefficients, the analytical model used for analyses, core parameters which affect the peak reactor pressure or the probability of fuel rod failure, and the results of the transient analyses. The NRC's acceptance criteria are based on GDC-28 for ensuring that the effects of postulated reactivity accidents do not result in damage to the RCPB greater than limited local yielding and do not cause sufficient damage to significantly impair the capability to cool the core. Specific review criteria are contained in SRP Section 15.4.8 and other guidance provided in Matrix 6 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

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

Regulatory Evaluation

Equipment malfunctions, operator errors, and abnormal occurrences could cause unplanned increases in reactor coolant inventory. Depending on the boron concentration and temperature of the injected water and the response of the automatic control systems, a power level increase may result and, without adequate controls, could lead to fuel damage or overpressurization of the RCS. Alternatively, a power level decrease and depressurization may result. Reactor protection and safety systems are actuated to mitigate these events. The NRC staff's review covers (1) the sequence of events, (2) the analytical model used for analyses, (3) the values of parameters used in the analytical model, and (4) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC-10 for the RCS being designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operations, including anticipated operational occurrences; (2) GDC-15 for the RCS and its associated auxiliaries being designed with appropriate margin to ensure that the RCPB will not be breached during normal operations, including anticipated operational occurrences; and (3) GDC-26 for the reliable control of reactivity changes to ensure that specified acceptable fuel design limits are not exceeded during normal operation, including anticipated operational occurrences. Specific review criteria are contained in SRP Section 15.5.1-2 and other guidance provided in Matrix 6 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.6.5.6. Decrease in Reactor Coolant Inventory

2.6.5.6.1. Inadvertent Opening of Pressurizer Pressure Relief Valve

Regulatory Evaluation

The inadvertent opening of a pressure relief valve results in a reactor coolant inventory decrease and a decrease in RCS pressure. A reactor trip normally occurs due to low RCS pressure. The NRC staff's review covers (1) the sequence of events, (2) the analytical model used for analyses, (3) the values of parameters used in the analytical model, and (4) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC-10 for the RCS being designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operations, including anticipated operational occurrences; (2) GDC-15 for the RCS and its associated auxiliaries being designed with appropriate margin to ensure that the RCPB will not be breached during normal operations, including anticipated operational occurrences; and (3) GDC-26 for the reactivity control systems to provide adequate control of reactivity changes to ensure that the acceptable fuel design limits are not exceeded during normal operations and anticipated transients during normal operations, including anticipated operational occurrences. Specific review criteria are contained in SRP Section 15.6.1 and other guidance provided in Matrix 6 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.6.5.6.2. Steam Generator Tube Rupture

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.6.5.6.3. Emergency Core Cooling System and Loss-of-Coolant Accidents

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.6.5.7. 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:

13. 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

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

The NRC staff's review is conducted to ensure that the above requirements are satisfied and that the setpoints for the ATWS mitigating system actuation circuitry (AMSAC) and DSS remain valid for the proposed EPU. In addition, for plants where a DSS is not specifically required by 10 CFR 50.62, the NRC staff verifies that the consequences of an ATWS are acceptable. The acceptance criteria is that the peak primary system pressure should not exceed the ASME Service Level C limit of 3200 psig. The peak ATWS pressure is primarily a function of the moderator temperature coefficient (MTC) and the primary system relief capacity. The NRC staff reviews (1) the limiting event determination, (2) the sequence of events, (3) the analytical model and its applicability, (4) the values of parameters used in the analytical model, and (5) the results of the analyses. If the licensee relies upon generic vendor analyses, the NRC staff reviews the licensee's justification of the applicability of those analyses to the plant under review and the operating conditions for the proposed EPU. Review guidance is provided in Matrix 6 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.6.6. Fuel Storage

2.6.6.1. New Fuel Storage

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.6.6.2. Spent Fuel Storage

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

[2.6.7. Additional Review Areas (Reactor Systems)]

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

INSERT 7

FOR

SECTION 3.3 - PWR TEMPLATE SAFETY EVALUATION

2.7 Source Terms and Radiological Consequences Analyses

2.7.1. Source Terms for Radwaste Systems Analyses

Regulatory Evaluation

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

NOTE: Use Sections 2.7.2 and 2.7.3 below if the licensee's radiological consequences analyses are based on an alternative source term.

2.7.2. Radiological Consequences Analyses Using Alternative Source Terms

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

Regulatory Evaluation

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

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

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

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

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

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

[2.7.3. Additional Review Areas (Radiological Consequences Analyses)]

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

NOTE: Use Sections 2.7.2 - 2.7.10 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)

2.7.2. Radiological Consequences of Main Steamline Failures Outside Containment

Regulatory Evaluation

The NRC staff reviews the analyses of the radiological consequences of a main steamline break (MSLB) outside the containment. The NRC staff's review includes (1) the sequence of events, models and assumptions used by the licensee for the calculation of the radiological doses; (2) evaluation of the TSs on the primary and secondary coolant iodine activities; and (3) determination of reactor coolant iodine concentration corresponding to a preaccident iodine spike and a concurrent iodine spike. The NRC's acceptance criteria for the radiological consequences of an MSLB outside containment are based on (1) GDC-19 for control room habitability and (2) 10 CFR Part 100 for the radiological consequences of a postulated accident. Specific review criteria are contained in SRP Sections 6.4 and 15.1.5.A, and other guidance provided in Matrix 7 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has evaluated the licensee's revised accident analyses for the radiological consequences of an MSLB outside containment and concludes that the licensee has adequately accounted for the effects of the proposed EPU on 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.

2.7.3. Radiological Consequences of a Reactor Coolant Pump Locked-Rotor Accident

Regulatory Evaluation

The NRC staff reviews the analyses of the radiological consequences of a reactor coolant pump locked-rotor accident. The review includes (1) determination of a need for a radiological consequences analysis; and (2) the sequence of events, models and assumptions used by the licensee for the calculation of radiological doses. The NRC's acceptance criteria for the radiological consequences of a reactor coolant pump locked-rotor accident are based on (1) GDC-19 for control room habitability and (2) 10 CFR Part 100 for the radiological consequences of a postulated accident. Specific review criteria are contained in SRP Sections 6.4 and 15.3.3-15.3.4; and other guidance provided in Matrix 7 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.7.4. Radiological Consequences of a Control Rod Ejection Accident

Regulatory Evaluation

The NRC staff reviews the analyses of the radiological consequences of a control rod ejection accident. The NRC staff's review includes the plant response to a control rod ejection accident and the calculation of radiological doses at the EAB and LPZ outer boundary and in the control room due to the releases resulting from a rod ejection accident. The purpose of the NRC staff's review is to (1) ensure that plant's procedures for recovery from a rod ejection accident and the plant's TSs are properly taken into account in computing the doses and (2) compare the calculated doses against the appropriate guidelines. The NRC's acceptance criteria for the radiological consequences of a control rod ejection accident are based on (1) GDC-19 for control room habitability and (2) 10 CFR Part 100 for mitigating the radiological consequences of an accident. Specific review criteria are contained in SRP Sections 6.4 and 15.4.8.A, and other guidance provided in Matrix 7 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

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

Regulatory Evaluation

The NRC staff reviews the analyses of the radiological consequences of failures outside the containment of small lines connected to the primary coolant pressure boundary (e.g., instrument lines and sample lines). The NRC staff's review includes (1) the identification of small lines postulated to fail and the isolation provisions for these lines; (2) the failure scenario; (3) the models and assumptions for the calculation of the radiological doses for the postulated failure; and (4) an evaluation of the primary coolant iodine activity, including the effects of a concurrent iodine spike, and the TSs for the reactor coolant iodine activity. The NRC's acceptance criteria for the radiological consequences of the failure of small lines carrying primary coolant outside containment are based on (1) GDC-19 for control room habitability and (2) GDC-55 for the isolation requirements of small-diameter lines connected to the primary system that are acceptable on the basis of meeting 10 CFR 100.11. Specific review criteria are contained in SRP Sections 6.4 and 15.6.2, and other guidance provided in Matrix 7 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.7.6. Radiological Consequences of Steam Generator Tube Rupture

Regulatory Evaluation

The NRC staff reviews the analysis of the radiological consequences of a postulated steam generator tube rupture (SGTR). The NRC staff's review includes (1) a review of the sequence of events and plant procedures for recovery from the accident to ensure that the most severe case of radioactive releases has been considered; (2) a review of the models and assumptions for the calculation of the radiological doses for the postulated accident; (3) an evaluation of the TSs on the primary and secondary coolant iodine activity concentration; and (4) an evaluation of the radiological consequences of an SGTR concurrent with a loss of offsite power and the most limiting single failure. The NRC staff's review includes two cases for the reactor coolant iodine concentration corresponding to a preaccident iodine spike and a concurrent iodine spike. The NRC's acceptance criteria for the radiological consequences of an SGTR are based on (1) GDC-19 for control room habitability and (2) 10 CFR Part 100 for mitigating the radiological consequences of an accident. Specific review criteria are contained in SRP Sections 6.4 and 15.6.3, and other guidance provided in Matrix 7 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.7.7. Radiological Consequences of a Design-Basis Loss-of-Coolant Accident

Regulatory Evaluation

The NRC staff reviews the analyses of the radiological consequences of a design-basis LOCA. The review includes a summary review of the doses from the hypothetical design-basis LOCA and a specific review of the doses from containment leakage and leakage from ESF components outside containment that contribute to the total LOCA doses. The NRC staff's review also includes (1) the methodology and results of calculations of the radiological consequences resulting from containment and ESF component leakage following a hypothetical LOCA; and (2) an assessment of the containment with respect to the assumptions and the input parameters for the dose calculations. The NRC staff's calculations are based on pertinent information in the safety analysis report (SAR) and considers the NRC staff's evaluation of dose-mitigating ESFs. The NRC's acceptance criteria for the radiological consequences of a design-basis LOCA are based on (1) GDC-19 for control room habitability and (2) 10 CFR Part 100 for mitigating the radiological consequences of an accident. Specific review criteria are contained in SRP Section 6.4 and Appendices A and B of SRP Section 15.6.5, and other guidance provided in Matrix 7 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.7.8. Radiological Consequences of Fuel Handling Accidents

Regulatory Evaluation

The NRC staff reviews the analyses of the radiological consequences of a postulated FHA. The purpose of this review is to evaluate the adequacy of system design features and plant procedures provided for the mitigation of the radiological consequences of accidents that involve damage to spent fuel. Such accidents include the dropping of a single fuel assembly and handling tool or a heavy object onto other spent fuel assemblies. Such accidents may occur inside the containment, along the fuel transfer canal, and in the fuel building. The NRC staff's review includes (1) the sequence of events, models, and assumptions used by the licensee for the calculation of radiological doses; (2) the adequacy of the ESFs provided for the purpose of mitigating potential accident doses; and (3) the containment ventilation system with respect to its function as a dose-mitigating ESF system, including the radiation detection system on the containment purge/vent lines for those plants that will vent or purge the containment during fuel handling operations. The NRC's acceptance criteria for the radiological consequences of FHAs are based on (1) GDC-19 for control room habitability; (2) GDC-61 for appropriate containment, confinement, and filtering systems; and (3) 10 CFR Part 100 for the calculated radiological consequences of FHAs. Specific review criteria are contained in SRP Sections 6.4 and 15.7.4, and other guidance provided in Matrix 7 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

2.7.9. Radiological Consequences of Spent Fuel Cask Drop Accidents

Regulatory Evaluation

The NRC staff reviews the analyses of the radiological consequences of the release of fission products from irradiated fuel in a spent fuel cask that is postulated to drop during cask handling operations. The NRC staff's review is conducted to verify various design and operations aspects of the system. The NRC staff's review includes (1) determining a need for a design-basis radiological analysis; (2) sequence of events, models and assumptions used by the licensee for the calculation of the radiological doses; and (3) comparing the calculated doses to exposure guidelines to determine the acceptability of the EAB and LPZ outer boundary distances and to confirm the adequacy of ESFs provided for the purpose of mitigating potential doses from spent fuel cask drop accidents, including the effects on control room habitablity. The NRC's acceptance criteria for the radiological consequences of spent fuel cask drop accidents are based on (1) GDC-19 for control room habitability; (2) GDC-61 for appropriate containment, confinement and filtering systems; and (3) 10 CFR Part 100 for the calculated radiological consequences of a spent fuel cask drop accident. Specific review criteria are contained in SRP Sections 6.4 and 15.7.5, and other guidance provided in Matrix 7 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

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

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

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SECTION 3.3 - PWR TEMPLATE SAFETY EVALUATION

2.8 Health Physics

2.8.1. Occupational and Public Radiation Doses

Regulatory Evaluation

The NRC staff conducts its review in this area to ascertain what overall effects the proposed EPU will have on both occupational and public radiation doses and to determine that the licensee has taken the necessary steps to ensure that any dose increases will be maintained as low as is reasonably achievable. The NRC staff's review includes an evaluation of any increases in radiation sources and how this may affect plant area dose rates, plant radiation zones, and plant area accessibility. The NRC staff evaluates how personnel doses needed to access plant vital areas following an accident are affected. The NRC staff considers the effects of the proposed EPU on nitrogen-16 levels in the plant and any effects this increase may have on radiation doses outside the plant and at the site boundary from skyshine. The NRC'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 8 of RS-001.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

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

[2.8.2. Additional Review Areas (Health Physics)]

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

INSERT 9

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SECTION 3.3 - PWR TEMPLATE SAFETY EVALUATION

2.9 Human Performance

2.9.1. Human Factors

Regulatory Evaluation

The area of human factors deals with programs, procedures, training, and plant design features related to operator performance during normal and accident conditions. The NRC staff's human factors evaluation is conducted to ensure that operator performance is not adversely affected as a result of system changes required for the proposed EPU. The NRC staff's review covers changes to operator actions, human-system interfaces, and procedures and training required for the proposed EPU. The NRC's acceptance criteria for human factors are based on GDC-19, 10 CFR 50.54(i) and (m), 10 CFR 50.59, 10 CFR 50.120, and 10 CFR 55.59. Specific review criteria are contained in SRP Sections 13.2.1, 13.2.2, 13.5.2.1, and 18.0.

Technical Evaluation

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

1. Changes in Emergency and Abnormal Operating Procedures

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

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

2. Changes to Operator Actions Sensitive to Power Uprate

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

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

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

3. Changes to Control Room Controls, Displays and Alarms

   Describe any changes the proposed EPU will have on the operator interfaces for control room controls, displays and alarms. For example, what zone markings (e.g. normal, marginal and out-of-tolerance ranges) on meters will change? What setpoints will change? How will the operators know of the change? Describe any controls, displays, alarms that will be upgraded from analog to digital instruments as a result of the proposed EPU and how operators were tested to determine they could use the instruments reliably. (SRP Section 18.0)

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

4. <u>Changes on the Safety Parameter Display System</u>

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

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

5. <u>Changes to the Operator Training Program and the Control Room Simulator</u>

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

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

<u>Conclusion</u>

The NRC staff has reviewed the changes to operator actions, human-system interfaces, procedures and training required for the proposed EPU and concludes that (1) the licensee has appropriately accounted for the effects of the proposed EPU on the available time for operator actions and (2) the licensee has taken appropriate actions to ensure that operators' performance is not adversely affected by the proposed EPU. The NRC staff further concludes that the licensee will continue to meet the requirements of 10 CFR 50.54(i) and (m), 10 CFR 50.59, 10 CFR 50.120, and 10 CFR 55.59 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.

[2.9.2. Additional Review Areas (Human Performance)]

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

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SECTION 3.3 - PWR TEMPLATE SAFETY EVALUATION

2.10 Power Ascension and Testing Plan

2.10.1. Approach to EPU Power Level and Test Plan

Regulatory Evaluation

The purpose of the power ascension and testing plan is to demonstrate that modifications to the plant are adequately designed and implemented and that the plant can be operated safely at the proposed EPU power level. The test program also provides additional assurance that the requested power uprate does not invalidate principle design criteria contained in the original licensing basis. The NRC staff's review includes an evaluation of: (1) plans for the initial approach to the proposed maximum licensed thermal power level, including verification of adequate plant performance, (2) transient testing requirements necessary to demonstrate that the plant can be operated safely 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 power ascension and testing plan are based on 10 CFR Part 50, Appendix B, Criterion XI, for the performance of all testing required to demonstrate that SSCs will perform satisfactorily in service. Specific review criteria are contained in SRP Section 14.2.1.

Technical Evaluation

[Insert technical evaluation.]

Conclusion

The NRC staff has reviewed the proposed EPU test plan related to initial approach to the proposed power level, steady state-performance, and transient testing, and concludes that the proposed plan will demonstrate that modifications made to the plant have been adequately designed and implemented and that the plant can be safely operated at the proposed power level. The NRC staff further concludes that the proposed EPU testing plan satisfies the requirements of 10 CFR Part 50, Appendix B, Criterion XI. Therefore, the NRC staff finds the proposed EPU test plan acceptable.

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

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

INSERT 11

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SECTION 3.3 - PWR TEMPLATE SAFETY EVALUATION

2.11. Risk Evaluation

2.11.1 Risk Evaluation of Extended Power Uprate

Regulatory Evaluation

A risk evaluation is conducted to (1) demonstrate that the risks associated with the proposed EPU are acceptable and (2) determine if “special circumstances” are created by the proposed EPU. As described in Appendix D of SRP Chapter 19, special circumstances are any issues that would potentially rebut the presumption of adequate protection provided by the licensee to meet the deterministic requirements and regulations. The NRC staff’s review covers the impact of the proposed EPU on core damage frequency (CDF) and large early release frequency (LERF) for the plant due to changes in the risks associated with internal events, external events, and shutdown operations. In addition, the NRC staff’s review covers the quality of the risk analyses used by the licensee to support the application for the proposed EPU. This includes a review of licensee actions to address issues or weaknesses that may have been raised in previous NRC staff reviews of the licensee’s individual plant examinations (IPEs) and individual plant examinations of external events (IPEEE), or by an industry peer review. The NRC’s risk acceptability guidelines are contained in RG 1.174. Specific review guidance is contained in Matrix 11 of RS-001 and its attachments.

Technical Evaluation

[Insert technical evaluation]

Conclusion

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

[2.11.2. Additional Review Areas (Risk Evaluation)]

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

SECTION 4

INSPECTION GUIDANCE

4.1 Inspection Requirements

Inspection Procedure (IP) 71004, "Power Uprates," describes the inspections required for power uprate related activities and provides guidance for the inspectors to use in conducting these inspections. In addition, the "Recommended Areas for Inspection" section of the final safety evaluation approving an EPU should be considered by inspectors when selecting a sample for implementing IP 71004. The recommendations in the final safety evaluation do not constitute inspection requirements, but are provided to give the inspectors insight into important bases for approving the EPU.