Contains proprietary information Per 10 CFR 2.390.
See Enclosure 1

TVA

Tennessee Valley Authority, Post Office Box 2000, Decatur, Alabama 35609-2000

February 23, 2005
TVA-BFN-TS-418

10 CFR 50.90

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop: OWFN, P1-35
Washington, D.C. 20555-0001

Gentlemen:

In the Matter of	)	Docket Nos. 50-260
Tennessee Valley Authority	)	50-296

BROWNS FERRY NUCLEAR PLANT (BFN) - UNITS 2, AND 3 - RESPONSE TO NRC's ACCEPTANCE REVIEW LETTER AND REQUEST FOR ADDITIONAL INFORMATION RELATED TO TECHNICAL SPECIFICATIONS (TS) CHANGE NO. TS-418 – REQUEST FOR EXTENDED POWER UPRATE OPERATION (TAC NOS MC3743 and MC3744)

This letter contains the additional information requested by the NRC Staff in its November 18, 2004 letter (Reference 1) that provided the results of the Staff's acceptance review of the BFN Units 2 and 3 Extended Power Uprate (EPU) application (BFN Units 2 and 3 Proposed Technical Specifications Change TS-418). TVA submitted TS-418 on June 25, 2004 (Reference 2), requesting a license amendment to increase the rated thermal power from 3458 megawatts thermal (MWt) to 3952 MWt, an approximate 15 percent increase in thermal power.

TS-418 was prepared based on the guidelines contained in General Electric (GE) Licensing Topical Reports (LTR) NEDC-32424P-A, "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate," (ELTR1) and NEDC-32523P-A, "Generic Evaluations for General Electric Boiling Water Reactor Extended Power Uprate," (ELTR2). By letter dated February 8, 1996, the NRC Staff concluded that the uprate program described in ELTR1

APO1

contains adequate scope and level of detail for review to address the effects of a power uprate on plant systems and components. NRC has previously reviewed and approved several other licensees that have submitted similar requests using the ELTR process.

The enclosures to this letter provide the additional information requested by NRC to consider TVA's application for EPU. Enclosure 1 to this letter provides TVA's response to the questions transmitted by Reference 1.

Enclosure 2 to this letter is the Extended Power Uprate RS-001 Revised Template Safety Evaluation. To aid the staff in preparing the BFN specific EPU Safety Evaluation, BFN has replaced the numeric values of the General Design Criteria (GDC) in the revised template safety evaluation to incorporate the draft GDC that corresponds to the Browns Ferry criteria that Browns Ferry was reviewed against during each unit's original licensing effort. These changes to the template are identified by change bars in the left margin. A cross reference between the 70 draft GDC and the current 64 GDC is included in Enclosure 2.

Enclosure 3 is the revised Extended Power Uprate RS-001 Areas of Review Matrix. The Matrix cross references the criteria in the NRC review standard with the information in the BFN Power Uprate Safety Analysis Report, the Framatome Uprate Safety Analysis Report, the BFN licensing basis with respect to the draft GDC and the approved ELTR for EPU. Notes have been added to the matrices to provide additional guidance to direct the reviewer to the specific safety analyses and conclusions.

Some of the information in Enclosure 1 is considered proprietary Framatome ANP (FANP). FANP requests that the proprietary information in the enclosure be withheld from public disclosure in accordance with 10 CFR 9.17(a)(4), 10 CFR 2.390(a)(4) and 10 CFR 2.390(b)(1). An affidavit supporting this request is included with Enclosure 1. A non-proprietary version is contained in Enclosure 4.

In a February 17, 2005 teleconference, the NRC Staff informed TVA that additional information would be required to support the justification for exception to large transient testing addressed in Enclosure 8 of the license amendment application, and Enclosures 1 and 4 TVA Reply 4 in this letter. TVA will provide this additional information by April 11, 2005. Additionally, as a result of the teleconference it is TVA's understanding that the information provided in the enclosures to this letter is sufficient for the NRC to accept the BFN Units 2 and 3 application and begin its review.

TVA is providing similar information regarding the Unit 1 EPU application in a separate submittal. Based on recent discussions with the NRC Staff to obtain clarification of the information requested, TVA was granted an additional seven days to respond to the information request. There are no new regulatory commitments associated with this submittal. If you have any questions concerning this letter, please telephone me at (256) 729-2636.

Pursuant to 28 U.S.G. § 1796 (1994), I declare under penalty of perjury that the forgoing is true and correct.

Executed on this 23rd day of February 23, 2005.

Sincerely,

T. E. Abney
Manager of Licensing
and Industry Affairs

Enclosures:

1. Reply to Results of Acceptance Review and Request for Additional Information for BFN Units 2 and 3 Extended Power Uprate Application (Proprietary Version)
2. Extended Power Uprate RS-001 Revised Template Safety Evaluation
3. Extended Power Uprate RS-001 Revised Areas of Review Matrix
4. Reply to Results of Acceptance Review and Request for Additional Information for BFN Units 2 and 3 Extended Power Uprate Application (Non-Proprietary Version)

References:

1. NRC letter, E. M. Hackett to K. W. Singer (TVA), "Browns Ferry Nuclear Plant, Units 2 and 3 – Results of Acceptance Review for Extended Power Uprate (TAC Nos. MC3743 and MC3744) (TS-418)," dated November 18, 2004.

2. TVA letter, T. E. Abney to NRC, "Browns Ferry Nuclear Plant (BFN) - Unit 1- Proposed Technical Specifications (TS) Change TS - 418 - Request for License Amendment - Extended Power Uprate (EPU) Operation," dated June 25, 2004.

Enclosure
cc (Enclosure):

U.S. Nuclear Regulatory Commission
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TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT UNITS 2 AND 3
DOCKET NOS. 50-260, 50-296

INDEX OF ENCLOSURES

Enclosure 1	Reply to Results of Acceptance Review and Request for Additional Information for BFN Units 2 and 3 Extended Power Uprate Application (Proprietary Version)
Enclosure 2	Extended Power Uprate RS-001 Revised Template Safety Evaluation
Enclosure 3	Extended Power Uprate RS-001 Revised Areas of Review Matrix
Enclosure 4	Reply to Results of Acceptance Review and Request for Additional Information for BFN Units 2 and 3 Extended Power Uprate Application (Non-Proprietary Version)

ENCLOSURE 1

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT UNITS 2 AND 3
DOCKET NOS. 50-260, AND 50-296

REPLY TO RESULTS OF ACCEPTANCE REVIEW AND REQUEST FOR ADDITIONAL INFORMATION FOR BFN UNITS 2 AND 3 EXTENDED POWER UPRATE APPLICATION
(Proprietary Version)

See Attached:

- Framatome ANP Affidavit
- Reply to Results of Acceptance Review And Request for Additional Information For BFN Units 2 and 3 Extended Power Uprate Application

AFFIDAVIT

STATE OF WASHINGTON)
) ss.
COUNTY OF BENTON)

1. My name is Jerald S. Holm. I am Manager, Product Licensing, for Framatome ANP, Inc. ("FANP"), and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by FANP to determine whether certain FANP information is proprietary. I am familiar with the policies established by FANP to ensure the proper application of these criteria.

3. I am familiar with the FANP attachment to the February 2005 TVA letter regarding TS-418 "*Browns Ferry Nuclear Plant (BFN) – Units 2, and 3 – Response To NRC's Acceptance Review Letter and Request for Additional Information Related to Technical Specifications (TS) Change No. TS-418 – Request For Extended Power Uprate Operation (TAC NOS MC3743 and MC3744)*", referred to herein as "Document." Information contained in this Document has been classified by FANP as proprietary in accordance with the policies established by FANP for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by FANP and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure.

6. The following criteria are customarily applied by FANP to determine whether information should be classified as proprietary:

(a) The information reveals details of FANP's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for FANP.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for FANP in product optimization or marketability.

(e) The information is vital to a competitive advantage held by FANP, would be helpful to competitors to FANP, and would likely cause substantial harm to the competitive position of FANP.

7. In accordance with FANP's policies governing the protection and control of information, proprietary information contained in this Document has been made available, on a limited basis, to others outside FANP only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. FANP policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

SUBSCRIBED before me this 17th day of February, 2005.

Susan K. McCoy
NOTARY PUBLIC, STATE OF WASHINGTON
MY COMMISSION EXPIRES: 1/10/08

SUSAN K. McCOY
COMMISSION EXPIRES
NOTARY
PUBLIC
1-10-08
STATE OF WASHINGTON

ENCLOSURE 2

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT UNITS 2 AND 3
DOCKET NOS. 50-260, 50-296

EXTENDED POWER UPRATE RS-001 REVISED TEMPLATE SAFETY EVALUATION

This enclosure is the revised Extended Power Uprate RS-001 Template Safety Evaluation. To aid the staff in preparing the BFN specific EPU Safety Evaluation, BFN has replaced the numeric values of the GDC in the revised template safety evaluation to incorporate the draft GDC that corresponds to the Browns Ferry criteria that Browns Ferry was reviewed against during each unit's original licensing effort. These changes to the template are identified by change bars in the left margin.

<u>See Attached:</u>

- AEC / GDC Matrix
- Extended Power Uprate RS-001 Revised Template Safety Evaluation

AEC / GDC Matrix		
AEC Draft GDC		10 CFR 50, Appendix A GDC
1		1
2		2
3		3
4		5
5		1
6		10
7		12
8		11
9		14
10		16
11		19
12		13
13		13
14		20
15		20
16		30
17		64
18		63
19		21
20		21
21		N/A
22		24
23		22
24		17
25		21
26		23
27		26
28		N/A
29		26
30		26
31		25
32		28
33		14 & 31
34		31
35		31
36		32
37		33 & 35
38		36 & 37
39		17 & 18
40		4

AEC / GDC Matrix		
AEC Draft GDC		10 CFR 50, Appendix A GDC
41		35 & 38
42		4
43		N/A
44		35
45		36
46		37
47		37
48		37
49		50
50		51
51		54 & 55
52		38
53		56 & 57
54		52
55		52
56		53
57		54
58		39
59		40
60		40
61		40
62		42
63		43
64		43
65		43
66		62
67		61 & 63
68		61
69		61
70		60

INSERT 1

FOR

SECTION 3.2 - BWR TEMPLATE SAFETY EVALUATION

2.1 Materials and Chemical Engineering

2.1.1 Reactor Vessel Material Surveillance Program

Regulatory Evaluation

The reactor vessel material surveillance program provides a means for determining and monitoring the fracture toughness of the reactor vessel beltline materials to support analyses for ensuring the structural integrity of the ferritic components of the reactor vessel. The NRC staff's review primarily focused on the effects of the proposed EPU on the licensee's reactor vessel surveillance capsule withdrawal schedule. The NRC's acceptance criteria are based on (1) draft General Design Criterion[1] (GDC)-9, insofar as it requires that the reactor coolant pressure boundary (RCPB) be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage; (2) draft GDC-33, insofar as it requires that the RCPB be capable of accommodating without rupture, and with only limited allowance for energy absorption through plastic deformation, the static and dynamic loads imposed on any boundary component as a result of any inadvertent and sudden release of energy to the coolant; (3) draft GDC-34 insofar as it requires that the RCPB be designed to minimize the probability of rapidly propagating type failures; (4) 10 CFR Part 50, Appendix H, which provides for monitoring changes in the fracture toughness properties of materials in the reactor vessel beltline region; and (5) 10 CFR 50.60, which requires compliance with the requirements of 10 CFR Part 50, Appendix H. Specific review criteria are contained in Standard Review Plan (SRP) Section 5.3.1 and other guidance provided in Matrix 1 of RS-001.

Technical Evaluation

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

Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed EPU on the reactor vessel surveillance withdrawal schedule and concludes that the licensee has adequately addressed changes in neutron fluence and their effects on the schedule. The NRC staff further concludes that the reactor vessel capsule withdrawal schedule is appropriate to ensure that the material surveillance program will continue to meet the requirements of 10 CFR Part 50, Appendix H, and 10 CFR 50.60, and will provide the licensee with information to ensure continued compliance with draft GDC-9, 33, and 34 in this respect following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the reactor vessel material surveillance program.

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[1] At the time of initial FSAR preparation, the design bases of each BFN unit were reevaluated against the draft of the 70 GDC current at the time of operating license application. UFSAR Appendix A documents the interpretations, discussions, and conclusions on how the design of the BFN plant conformed to the draft GDCs.

2.1.2 Pressure-Temperature Limits and Upper-Shelf Energy

Regulatory Evaluation

Pressure-temperature (P-T) limits are established to ensure the structural integrity of the ferritic components of the RCPB during any condition of normal operation, including anticipated operational occurrences and hydrostatic tests. The NRC staff's review of P-T limits covered the P-T limits methodology and the calculations for the number of effective full power years specified for the proposed EPU, considering neutron embrittlement effects and using linear elastic fracture mechanics. The NRC's acceptance criteria for P-T limits are based on (1) draft GDC-9, insofar as it requires that the RCPB be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage; (2) draft GDC-33, insofar as it requires that the RCPB be capable of accommodating without rupture, and with only limited allowance for energy absorption through plastic deformation, the static and dynamic loads imposed on any boundary component as a result of any inadvertent and sudden release of energy to the coolant; (3) draft GDC-34 insofar as it requires that the RCPB be designed to minimize the probability of rapidly propagating type failures; (4) draft GDC-35 insofar as it requires that service temperatures for RCPB components constructed of ferritic materials ensure the structural integrity of such components when subjected to potential loadings; (5)10 CFR Part 50, Appendix G, which specifies fracture toughness requirements for ferritic components of the RCPB; and (6) 10 CFR 50.60, which requires compliance with the requirements of 10 CFR Part 50, Appendix G. Specific review criteria are contained in SRP Section 5.3.2 and other guidance provided in Matrix 1 of RS-001.

Technical Evaluation

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

Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed EPU on the P-T limits for the plant and concludes that the licensee has adequately addressed changes in neutron fluence and their effects on the P-T limits. The NRC staff further concludes that the licensee has demonstrated the validity of the proposed P-T limits for operation under the proposed EPU conditions. Based on this, the NRC staff concludes that the proposed P-T limits will continue to meet the requirements of 10 CFR Part 50, Appendix G, and 10 CFR 50.60 and will enable the licensee to comply with draft GDC-9, 33, 34, and 35 in this respect following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the proposed P-T limits.

2.1.3 Reactor Internal and Core Support Materials

Regulatory Evaluation

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

Technical Evaluation

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

Conclusion

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

2.1.4 Reactor Coolant Pressure Boundary Materials

Regulatory Evaluation

The RCPB defines the boundary of systems and components containing the high-pressure fluids produced in the reactor. The NRC staff's review of RCPB materials covered their specifications, compatibility with the reactor coolant, fabrication and processing, susceptibility to degradation, and degradation management programs. The NRC's acceptance criteria for RCPB materials are based on (1) 10 CFR 50.55a and draft GDC-1, insofar as they require that SSCs important to safety be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety functions to be performed; (2) draft GDC-40 and 42, insofar as they require that protection be provided for engineered safety features (ESFs) against the dynamic effects and missiles that might result from plant equipment failures, as well as the effects of a loss-of-coolant accident; (3) draft GDC-9 and 33, insofar as they require that the RCPB be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage; (4) draft GDC-34 insofar as it requires that the RCPB be designed to minimize the probability of rapidly propagating type failures; (5) draft GDC-35 insofar as it requires that service temperatures for RCPB components constructed of ferritic materials ensure the structural integrity of such components when subjected to potential loadings; and (6) 10 CFR Part 50, Appendix G, which specifies fracture toughness requirements for ferritic components of the RCPB. Specific review criteria are contained in SRP Section 5.2.3 and other guidance provided in Matrix 1 of RS-001. Additional review guidance for primary water stress-corrosion cracking (PWSCC) of dissimilar metal welds and associated inspection programs is contained in Generic Letter (GL) 97-01, Information Notice (IN) 00-17, Bulletin (BL) 01-01, BL 02-01, and BL 02-02. Additional review guidance for thermal embrittlement of cast austenitic stainless steel components is contained in a letter from C. Grimes, NRC, to D. Walters, Nuclear Energy Institute (NEI), dated May 19, 2000.

Technical Evaluation

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

Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed EPU on the susceptibility of RCPB materials to known degradation mechanisms and concludes that the licensee has identified appropriate degradation management programs to address the effects of changes in system operating temperature on the integrity of RCPB materials. The NRC staff further concludes that the licensee has demonstrated that the RCPB materials will continue to be acceptable following implementation of the proposed EPU and will continue to meet the requirements of draft GDC-1, 9, 33, 34, 35, 40, and 42, 10 CFR Part 50, Appendix G, and 10 CFR 50.55a. Therefore, the NRC staff finds the proposed EPU acceptable with respect to RCPB materials.

2.1.5 Protective Coating Systems (Paints) - Organic Materials

Regulatory Evaluation

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

Technical Evaluation

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

Conclusion

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

2.1.6 Flow-Accelerated Corrosion

Regulatory Evaluation

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

Technical Evaluation

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

Conclusions

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

2.1.7 Reactor Water Cleanup System

Regulatory Evaluation

The reactor water cleanup system (RWCS) provides a means for maintaining reactor water quality by filtration and ion exchange and a path for removal of reactor coolant when necessary. Portions of the RWCS comprise the RCPB. The NRC staff's review of the RWCS included component design parameters for flow, temperature, pressure, heat removal capability, and impurity removal capability; and the instrumentation and process controls for proper system operation and isolation. The review consisted of evaluating the adequacy of the plant's TSs in these areas under the proposed EPU conditions. The NRC's acceptance criteria for the RWCS are based on (1) draft GDC-9, insofar as it requires that the RCPB be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage; (2) draft GDC-70, insofar as it requires that the plant design include means to control the release of radioactive effluents; and (3) draft GDC-51, insofar as it requires that systems that contain radioactivity be designed with appropriate confinement. Specific review criteria are contained in SRP Section 5.4.8.

Technical Evaluation

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

Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed EPU on the RWCS and concludes that the licensee has adequately addressed changes in impurity levels and pressure and their effects on the RWCS. The NRC staff further concludes that the licensee has demonstrated that the RWCS will continue to be acceptable following implementation of the proposed EPU and will continue to meet the requirements of draft GDC-9, 51, and 70. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the RWCS.

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

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

INSERT 2

FOR

SECTION 3.2 - BWR TEMPLATE SAFETY EVALUATION

2.2 Mechanical and Civil Engineering

2.2.1 Pipe Rupture Locations and Associated Dynamic Effects

Regulatory Evaluation

SSCs important to safety could be impacted by the pipe-whip dynamic effects of a pipe rupture. The NRC staff conducted a review of pipe rupture analyses to ensure that SSCs important to safety are adequately protected from the effects of pipe ruptures. The NRC staff's review covered (1) the implementation of criteria for defining pipe break and crack locations and configurations, (2) the implementation of criteria dealing with special features, such as augmented inservice inspection (ISI) programs or the use of special protective devices such as pipe-whip restraints, (3) pipe-whip dynamic analyses and results, including the jet thrust and impingement forcing functions and pipe-whip dynamic effects, and (4) the design adequacy of supports for SSCs provided to ensure that the intended design functions of the SSCs will not be impaired to an unacceptable level as a result of pipe-whip or jet impingement loadings. The NRC staff's review focused on the effects that the proposed EPU may have on items (1) thru (4) above. The NRC's acceptance criteria are based on draft GDC-40 insofar as it requires that protection be provided for ESFs against the dynamic effects and missiles that might result from plant equipment failures. Specific review criteria are contained in SRP Section 3.6.2.

Technical Evaluation

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

Conclusion

The NRC staff has reviewed the licensee's evaluations related to determinations of rupture locations and associated dynamic effects and concludes that the licensee has adequately addressed the effects of the proposed EPU on them. The NRC staff further concludes that the licensee has demonstrated that ESFs will continue to meet the requirements of draft GDC-40 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the determination of rupture locations and dynamic effects associated with the postulated rupture of piping.

2.2.2 Pressure-Retaining Components and Component Supports

Regulatory Evaluation

The NRC staff has reviewed the structural integrity of pressure-retaining components (and their supports) designed in accordance with the American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (B&PV Code), Section III, Division 1, and draft GDC 1, 2, 9, 33, 40 and 42. The NRC staff's review focused on the effects of the proposed EPU on the design input parameters and the design-basis loads and load combinations for normal operating, upset, emergency, and faulted conditions. The NRC staff's review covered (1) the analyses of flow-induced vibration and (2) the analytical methodologies, assumptions, ASME Code editions, and computer programs used for these analyses. The NRC staff's review also included a comparison of the resulting stresses and cumulative fatigue usage factors (CUFs) against the code-allowable limits. The NRC's acceptance criteria are based on (1) 10 CFR 50.55a and draft GDC-1, insofar as they require that those systems and components which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety functions to be performed; (2) draft GDC-2, insofar as it requires that those systems and components which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences be designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions; (3) draft GDC-40 and 42, insofar as they require that protection be provide for ESFs against the dynamic effects that might result from plant equipment failures, as well as the effects of a loss of coolant accident; and (4) draft GDC-9 and 33, insofar as they require that the RCPB be designed and constructed so as to have an exceedingly low probability of RCPB gross rupture or significant leakage; and (5) draft GDC-34 insofar as it requires that the RCPB be designed to minimize the probability of rapidly propagating type failures. Specific review criteria are contained in SRP Sections 3.9.1, 3.9.2, 3.9.3, and 5.2.1.1; and other guidance provided in Matrix 2 of RS-001.

Technical Evaluation

Nuclear Steam Supply System Piping, Components, and Supports

[Insert technical evaluation for nuclear steam supply system (NSSS) piping, components, and supports. Include an intermediate conclusion in the form of "Because [summarize reasons], the NSSS piping, components, and supports are adequate under the proposed EPU conditions."]

Balance-of-Plant Piping, Components, and Supports

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

Reactor Vessel and Supports

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

Control Rod Drive Mechanism

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

Recirculation Pumps and Supports

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

Conclusion

The NRC staff has reviewed the licensee's evaluations related to the structural integrity of pressure-retaining components and their supports. For the reasons set forth above, the NRC staff concludes that the licensee has adequately addressed the effects of the proposed EPU on these components and their supports. Based on the above, the NRC staff further concludes that the licensee has demonstrated that pressure-retaining components and their supports will continue to meet the requirements of 10 CFR 50.55a, draft GDC-1, 2, 9, 33, 34, 40, and 42 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the structural integrity of the pressure-retaining components and their supports.

2.2.3 Reactor Pressure Vessel Internals and Core Supports

Regulatory Evaluation

Reactor pressure vessel internals consist of all the structural and mechanical elements inside the reactor vessel, including core support structures. The NRC staff reviewed the effects of the proposed EPU on the design input parameters and the design-basis loads and load combinations for the reactor internals for normal operation, upset, emergency, and faulted conditions. These include pressure differences and thermal effects for normal operation, transient pressure loads associated with loss-of-coolant accidents (LOCAs), and the identification of design transient occurrences. The NRC staff's review covered (1) the analyses of flow-induced vibration for safety-related and non-safety-related reactor internal components and (2) the analytical methodologies, assumptions, ASME Code editions, and computer programs used for these analyses. The NRC staff's review also included a comparison of the resulting stresses and CUFs against the corresponding Code-allowable limits. The NRC's acceptance criteria are based on (1) 10 CFR 50.55a and draft GDC-1, insofar as they require that those systems and components which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety functions to be performed; (2) draft GDC-2, insofar as it requires that those systems and components which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences be designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions; (3) draft GDC-40 and 42, insofar as they require that protection be provided for ESFs against the dynamic effects and missiles that might result from plant equipment failures, as well as the effects of a loss of coolant accident; and (4) draft GDC-6, insofar as it requires that the reactor core be designed with appropriate margin to assure that acceptable fuel damage limits (AFDLs) are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences. Specific review criteria are contained in SRP Sections 3.9.1, 3.9.2, 3.9.3, and 3.9.5; and other guidance provided in Matrix 2 of RS-001.

Technical Evaluation

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

Conclusion

The NRC staff has reviewed the licensee's evaluations related to the structural integrity of reactor internals and core supports and concludes that the licensee has adequately addressed the effects of the proposed EPU on the reactor internals and core supports. The NRC staff further concludes that the licensee has demonstrated that the reactor internals and core supports will continue to meet the requirements of 10 CFR 50.55a, draft GDC-1, 2, 6, 40, and 42 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the design of the reactor internal and core supports.

2.2.4 Safety-Related Valves and Pumps

Regulatory Evaluation

The NRC's staff's review included certain safety-related pumps and valves typically designated as Class 1, 2, or 3 under Section III of the ASME B&PV Code and within the scope of Section XI of the ASME B&PV Code and the ASME Operations and Maintenance (O&M) Code, as applicable. The NRC staff's review focused on the effects of the proposed EPU on the required functional performance of the valves and pumps. The review also covered any impacts that the proposed EPU may have on the licensee's motor-operated valve (MOV) programs related to GL 89-10, GL 96-05, and GL 95-07. The NRC staff also evaluated the licensee's consideration of lessons learned from the MOV program and the application of those lessons learned to other safety-related power-operated valves. The NRC's acceptance criteria are based on (1) draft GDC-1, insofar as it requires that those systems and components which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed; (2) draft GDC-38, 46, 47, 48, 59, 60, 61, 63, 64, and 65 insofar as they require that the emergency core cooling system (ECCS), the containment heat removal system, the containment atomospheric cleanup systems, and the cooling water system, respectively, be designed to permit appropriate periodic testing to ensure the leak-tight integrity and performance of their active components; (3) draft GDC-57, insofar as it requires that piping systems penetrating containment be designed with the capability to periodically test the operability of the isolation valves to determine if valve leakage is within acceptable limits; and (4) 10 CFR 50.55a(f), insofar as it requires that pumps and valves subject to that section must meet the inservice testing program requirements identified in that section. Specific review criteria are contained in SRP Sections 3.9.3 and 3.9.6; and other guidance provided in Matrix 2 of RS-001.

Technical Evaluation

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

Conclusion

The NRC staff has reviewed the licensee's assessments related to the functional performance of safety-related valves and pumps and concludes that the licensee has adequately addressed the effects of the proposed EPU on safety-related pumps and valves. The NRC staff further concludes that the licensee has adequately evaluated the effects of the proposed EPU on its MOV programs related to GL 89-10, GL 96-05, and GL 95-07, and the lessons learned from those programs to other safety-related, power-operated valves. Based on this, the NRC staff concludes that the licensee has demonstrated that safety-related valves and pumps will continue to meet the requirements of draft GDC-1, 38, 46, 47, 48, 57, 59, 60, 61, 63, 64, and 65, and 10 CFR 50.55a(f) following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to safety-related valves and pumps.

2.2.5 Seismic and Dynamic Qualification of Mechanical and Electrical Equipment

Regulatory Evaluation

Mechanical and electrical equipment covered by this section includes equipment associated with systems that are essential to emergency reactor shutdown, containment isolation, reactor core cooling, and containment and reactor heat removal. Equipment associated with systems essential to preventing significant releases of radioactive materials to the environment are also covered by this section. The NRC staff's review focused on the effects of the proposed EPU on the qualification of the equipment to withstand seismic events and the dynamic effects associated pipe-whip and jet impingement forces. The primary input motions due to the safe shutdown earthquake (SSE) are not affected by an EPU. The NRC's acceptance criteria are based on (1) draft GDC-1, insofar as it requires that that those systems and components which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed; (2) draft GDC-2, insofar as it requires that those systems and components which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences be designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions; (3) 10 CFR Part 100, Appendix A, which sets forth the principal seismic and geologic considerations for the evaluation of the suitability of plant design bases established in consideration of the seismic and geologic characteristics of the plant site; (4) draft GDC-40 and 42, insofar as they require that protection be provided for ESFs against the dynamic effects and missiles that might result from plant equipment failures, as well as the effects of a loss of coolant accident; (5) draft GDC-9 and 33, insofar as they require that the RCPB be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage; (6) draft GDC-34 insofar as it requires that the RCPB be designed to minimize the probability of rapidly propagating type failures; and (7) 10 CFR Part 50, Appendix B, which sets quality assurance requirements for safety-related equipment. Specific review criteria are contained in SRP Section 3.10.

Technical Evaluation

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

Conclusion

The NRC staff has reviewed the licensee's evaluations of the effects of the proposed EPU on the qualification of mechanical and electrical equipment and concludes that the licensee has (1) adequately addressed the effects of the proposed EPU on this equipment and (2) demonstrated that the equipment will continue to meet the requirements of draft GDC-1, 2, 9, 33, 34, 40, and 42; 10 CFR Part 100, Appendix A; and 10 CFR Part 50, Appendix B, following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the qualification of the mechanical and electrical equipment.

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

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

INSERT 3

FOR

SECTION 3.2 - BWR TEMPLATE SAFETY EVALUATION

2.3 Electrical Engineering

2.3.1 Environmental Qualification of Electrical Equipment

Regulatory Evaluation

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

Technical Evaluation

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

Conclusion

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

2.3.2 Offsite Power System

Regulatory Evaluation

The offsite power system includes two or more physically independent circuits capable of operating independently of the onsite standby power sources. The NRC staff's review covered the descriptive information, analyses, and referenced documents for the offsite power system; and the stability studies for the electrical transmission grid. The NRC staff's review focused on whether the loss of the nuclear unit, the largest operating unit on the grid, or the most critical transmission line will result in the loss of offsite power (LOOP) to the plant following implementation of the proposed EPU. The NRC's acceptance criteria for offsite power systems are based on draft GDC-39. Specific review criteria are contained in SRP Sections 8.1 and 8.2, Appendix A to SRP Section 8.2, and Branch Technical Positions (BTPs) PSB-1 and ICSB-11.

Technical Evaluation

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

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the offsite power system and concludes that the offsite power system will continue to meet the requirements of draft GDC-39 following implementation of the proposed EPU. Adequate physical and electrical separation exists and the offsite power system has the capacity and capability to supply power to all safety loads and other required equipment. The NRC staff further concludes that the impact of the proposed EPU on grid stability is insignificant. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the offsite power system.

2.3.3 AC Onsite Power System

Regulatory Evaluation

The alternating current (ac) onsite power system includes those standby power sources, distribution systems, and auxiliary supporting systems provided to supply power to safety-related equipment. The NRC staff's review covered the descriptive information, analyses, and referenced documents for the ac onsite power system. The NRC's acceptance criteria for the ac onsite power system are based on draft GDC-24 and 39, insofar as they require the system to have the capacity and capability to perform its intended functions during anticipated operational occurrences and accident conditions. Specific review criteria are contained in SRP Sections 8.1 and 8.3.1.

Technical Evaluation

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

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the ac onsite power system and concludes that the licensee has adequately accounted for the effects of the proposed EPU on the system's functional design. The NRC staff further concludes that the ac onsite power system will continue to meet the requirements of draft GDC-24 and 39 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the ac onsite power system.

2.3.4 DC Onsite Power System

Regulatory Evaluation

The direct current (dc) onsite power system includes the dc power sources and their distribution and auxiliary supporting systems that are provided to supply motive or control power to safety-related equipment. The NRC staff's review covered the information, analyses, and referenced documents for the dc onsite power system. The NRC's acceptance criteria for the dc onsite power system are based on draft GDC-24 and 39, insofar as they require the system to have the capacity and capability to perform its intended functions during anticipated operational occurrences and accident conditions. Specific review criteria are contained in SRP Sections 8.1 and 8.3.2

Technical Evaluation

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

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the dc onsite power system and concludes that the licensee has adequately accounted for the effects of the proposed EPU on the system's functional design. The NRC staff further concludes that the dc onsite power system will continue to meet the requirements of draft GDC-24 and 39 following implementation of the proposed EPU. Adequate physical and electrical separation exists and the system has the capacity and capability to supply power to all safety loads and other required equipment. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the dc onsite power system.

2.3.5 Station Blackout

Regulatory Evaluation

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

Technical Evaluation

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

Conclusion

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

[2.3.6 Additional Review Areas (Electrical Engineering)]

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

INSERT 4

FOR

SECTION 3.2 - BWR TEMPLATE SAFETY EVALUATION

2.4 Instrumentation and Controls

2.4.1 Reactor Protection, Safety Features Actuation, and Control Systems

Regulatory Evaluation

Instrumentation and control systems are provided (1) to control plant processes having a significant impact on plant safety, (2) to initiate the reactivity control system (including control rods), (3) to initiate the engineered safety features (ESF) systems and essential auxiliary supporting systems, and (4) for use to achieve and maintain a safe shutdown condition of the plant. Diverse instrumentation and control systems and equipment are provided for the express purpose of protecting against potential common-mode failures of instrumentation and control protection systems. The NRC staff conducted a review of the reactor trip system, engineered safety feature actuation system (ESFAS), safe shutdown systems, control systems, and diverse instrumentation and control systems for the proposed EPU to ensure that the systems and any changes necessary for the proposed EPU are adequately designed such that the systems continue to meet their safety functions. The NRC staff's review was also conducted to ensure that failures of the systems do not affect safety functions. The NRC's acceptance criteria related to the quality of design of protection and control systems are based on 10 CFR 50.55a(a)(1), 10 CFR 50.55a(h), and draft GDC-1, 11, 12, 14, 15, 19, 20, 22, 23, 25, 26, 40, and 42. Specific review criteria are contained in SRP Sections 7.0, 7.2, 7.3, 7.4, 7.7, and 7.8.

Technical Evaluation

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

Conclusion

The NRC staff has reviewed the licensee's application related to the effects of the proposed EPU on the functional design of the reactor trip system, ESFAS, safe shutdown system, and control systems. The NRC staff concludes that the licensee has adequately addressed the effects of the proposed EPU on these systems and that the changes that are necessary to achieve the proposed EPU are consistent with the plant's design basis. The NRC staff further concludes that the systems will continue to meet the requirements of 10 CFR 50.55a(a)(1), 10 CFR 50.55(a)(h), and draft GDC-1, 11, 12, 14, 15, 19, 20, 22, 23, 25, 26, 40, and 42. 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 <u>Plant Systems</u>

2.5.1 <u>Internal Hazards</u>

2.5.1.1 Flooding

2.5.1.1.1 Flood Protection

<u>Regulatory Evaluation</u>

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

<u>Technical Evaluation</u>

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

<u>Conclusion</u>

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 draft GDC-2 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to flood protection.

2.5.1.1.2 Equipment and Floor Drains

Regulatory Evaluation

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

Technical Evaluation

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

Conclusion

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

2.5.1.1.3 Circulating Water System

Regulatory Evaluation

The circulating water system (CWS) provides a continuous supply of cooling water to the main condenser to remove the heat rejected by the turbine cycle and auxiliary systems. The NRC staff's review of the CWS focused on changes in flooding analyses that are necessary due to increases in fluid volumes or installation of larger capacity pumps or piping needed to accommodate the proposed EPU. The NRC's acceptance criteria for the CWS are based on draft GDC-40 for the effects of flooding of safety-related areas due to leakage from the CWS and the effects of malfunction or failure of a component or piping of the CWS on the functional performance capabilities of safety-related SSCs. Specific review criteria are contained in SRP Section 10.4.5.

Technical Evaluation

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

Conclusion

The NRC staff has reviewed the licensee's assessment of the modifications to the CWS and concludes that the licensee has adequately evaluated these modifications. The NRC staff concludes that, consistent with the requirements of draft GDC-40, the increased volumes of fluid leakage that could potentially result from these modifications would not result in the failure of safety-related SSCs following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the CWS.

2.5.1.2 Missile Protection

2.5.1.2.1. Internally Generated Missiles

Regulatory Evaluation

The NRC staff's review concerns missiles that could result from in-plant component overspeed failures and high-pressure system ruptures. The NRC staff's review of potential missile sources covered pressurized components and systems, and high-speed rotating machinery. The NRC staff's review was conducted to ensure that safety-related SSCs are adequately protected from internally generated missiles. In addition, for cases where safety-related SSCs are located in areas containing non-safety-related SSCs, the NRC staff reviewed the non-safety-related SSCs to ensure that their failure will not preclude the intended safety function of the safety-related SSCs. The NRC staff's review focused on any increases in system pressures or component overspeed conditions that could result during plant operation, anticipated operational occurrences, or changes in existing system configurations such that missile barrier considerations could be affected. The NRC's acceptance criteria for the protection of SSCs important to safety against the effects of internally generated missiles that may result from equipment failures are based on draft GDC-40. Specific review criteria are contained in SRP Sections 3.5.1.1 and 3.5.1.2.

Technical Evaluation

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

Conclusion

The NRC staff has reviewed the changes in system pressures and configurations that are required for the proposed EPU and concludes that SSCs important to safety will continue to be protected from internally generated missiles and will continue to meet the requirements of draft GDC-40 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to internally generated missiles.

2.5.1.2.2 Turbine Generator

Regulatory Evaluation

The turbine control system, steam inlet stop and control valves, low pressure turbine steam intercept and inlet control valves, and extraction steam control valves control the speed of the turbine under normal and abnormal conditions, and are thus related to the overall safe operation of the plant. The NRC staff's review of the turbine generator focused on the effects of the proposed EPU on the turbine overspeed protection features to ensure that a turbine overspeed condition above the design overspeed is very unlikely. The NRC's acceptance criteria for the turbine generator are based on draft GDC-40, and relates to protection of ESFs from the effects of turbine missiles by providing a turbine overspeed protection system (with suitable redundancy) to minimize the probability of generating turbine missiles. Specific review criteria are contained in SRP Section 10.2.

Technical Evaluation

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

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the turbine generator and concludes that the licensee has adequately accounted for the effects of changes in plant conditions on turbine overspeed. The NRC staff concludes that the turbine generator will continue to provide adequate turbine overspeed protection to minimize the probability of generating turbine missiles and will continue to meet the requirements of draft GDC-40 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the turbine generator.

2.5.1.3 Pipe Failures

Regulatory Evaluation

The NRC staff conducted a review of the plant design for protection from piping failures outside containment to ensure that (1) such failures would not cause the loss of needed functions of safety-related systems and (2) the plant could be safely shut down in the event of such failures. The NRC staff's review of pipe failures included high and moderate energy fluid system piping located outside of containment. The NRC staff's review focused on the effects of pipe failures on plant environmental conditions, control room habitability, and access to areas important to safe control of postaccident operations where the consequences are not bounded by previous analyses. The NRC's acceptance criteria for pipe failures are based on draft GDC-40 and 42, insofar that they require that ESFs be designed to accommodate the dynamic effects of postulated pipe ruptures, as well as the effects of a loss of coolant accident. Specific review criteria are contained in SRP Section 3.6.1.

Technical Evaluation

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

Conclusion

The NRC staff has reviewed the changes that are necessary for the proposed EPU and the licensee's proposed operation of the plant, and concludes that SSCs important to safety will continue to be protected from the dynamic effects of postulated piping failures in fluid systems outside containment and will continue to meet the requirements of draft GDC-40 and 42 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to protection against postulated piping failures in fluid systems outside containment.

2.5.1.4 Fire Protection

Regulatory Evaluation

The purpose of the fire protection program (FPP) is to provide assurance, through a defense-in-depth design, that a fire will not prevent the performance of necessary safe plant shutdown functions and will not significantly increase the risk of radioactive releases to the environment. The NRC staff's review focused on the effects of the increased decay heat on the plant's safe shutdown analysis to ensure that SSCs required for the safe shutdown of the plant are protected from the effects of the fire and will continue to be able to achieve and maintain safe shutdown following a fire. The NRC's acceptance criteria for the FPP are based on (1) 10 CFR 50.48 and associated Appendix R to 10 CFR Part 50, insofar as they require the development of an FPP to ensure, among other things, the capability to safely shut down the plant; (2) draft GDC-3, insofar as it requires that the reactor facility be designed (a) to minimize the probability of events, such as fire and explosions, and (b) to minimize the potential effects of such events to safety; and (3) draft GDC-4, insofar as it requires that reactor facilities shall not share systems or components unless it is shown safety is not impaired by the sharing. Specific review criteria are contained in SRP Section 9.5.1, as supplemented by the guidance provided in Attachment 2 to Matrix 5 of Section 2.1 of RS-001.

Technical Evaluation

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

Conclusion

The NRC staff has reviewed the licensee's fire-related safe shutdown assessment and concludes that the licensee has adequately accounted for the effects of the increased decay heat on the ability of the required systems to achieve and maintain safe shutdown conditions. The NRC staff further concludes that the FPP will continue to meet the requirements of 10 CFR 50.48, Appendix R to 10 CFR Part 50, and draft GDC-3 and 4 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to fire protection.

2.5.2 Fission Product Control

2.5.2.1 Fission Product Control Systems and Structures

Regulatory Evaluation

The NRC staff's review for fission product control systems and structures covered the basis for developing the mathematical model for DBLOCA dose computations, the values of key parameters, the applicability of important modeling assumptions, and the functional capability of ventilation systems used to control fission product releases. The NRC staff's review primarily focused on any adverse effects that the proposed EPU may have on the assumptions used in the analyses for control of fission products. The NRC's acceptance criteria are based on draft GDC-70, insofar as it requires that the facility design include those means necessary to maintain radioactivity control on the basis of 10CFR50.67 dose guidelines for potential reactor accidents. Specific review criteria are contained in SRP Section 6.5.3.

Technical Evaluation

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

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on fission product control systems and structures. The NRC staff concludes that the licensee has adequately accounted for the increase in fission products and changes in expected environmental conditions that would result from the proposed EPU. The NRC staff further concludes that the fission product control systems and structures will continue to provide adequate fission product removal in postaccident environments following implementation of the proposed EPU. Based on this, the NRC staff also concludes that the fission product control systems and structures will continue to meet the requirements of draft GDC-70. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the fission product control systems and structures.

2.5.2.2 Main Condenser Evacuation System

Regulatory Evaluation

The main condenser evacuation system (MCES) generally consists of two subsystems: (1) the "hogging" or startup system which initially establishes main condenser vacuum and (2) the system which maintains condenser vacuum once it has been established. The NRC staff's review focused on modifications to the system that may affect gaseous radioactive material handling and release assumptions, and design features to preclude the possibility of an explosion (if the potential for explosive mixtures exists). The NRC's acceptance criteria for the MCES are based on (1) draft GDC-70, insofar as it requires that the plant design include means to control the release of radioactive effluents; and (2) draft GDC-17, insofar as it requires that means be provided for monitoring effluent discharge paths and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences and postulated accidents. Specific review criteria are contained in SRP Section 10.4.2.

Technical Evaluation

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

Conclusion

The NRC staff has reviewed the licensee's assessment of required changes to the MCES and concludes that the licensee has adequately evaluated these changes. The NRC staff concludes that the MCES will continue to maintain its ability to control and provide monitoring for releases of radioactive materials to the environment following implementation of the proposed EPU. The NRC also concludes that the MCES will continue meet the requirements of draft GDC-17 and 70. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the MCES.

2.5.2.3 Turbine Gland Sealing System

Regulatory Evaluation

The turbine gland sealing system is provided to control the release of radioactive material from steam in the turbine to the environment. The NRC staff reviewed changes to the turbine gland sealing system with respect to factors that may affect gaseous radioactive material handling (e.g., source of sealing steam, system interfaces, and potential leakage paths). The NRC's acceptance criteria for the turbine gland sealing system are based on (1) draft GDC-70, insofar as it requires that the plant design include means to control the release of radioactive effluents; and (2) draft GDC-17, insofar as it requires that means be provided for monitoring effluent discharge paths and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences and postulated accidents. Specific review criteria are contained in SRP Section 10.4.3.

Technical Evaluation

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

Conclusion

The NRC staff has reviewed the licensee's assessment of required changes to the turbine gland sealing system and concludes that the licensee has adequately evaluated these changes. The NRC staff concludes that the turbine gland sealing system will continue to maintain its ability to control and provide monitoring for releases of radioactive materials to the environment consistent with draft GDC-17 and 70. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the turbine gland sealing system.

2.5.2.4 Main Steam Isolation Valve Leakage Control System

[Not applicable. BFN does not have a MSIV leakage control system.]

Regulatory Evaluation

Technical Evaluation

Conclusion

2.5.3 Component Cooling and Decay Heat Removal

2.5.3.1 Spent Fuel Pool Cooling and Cleanup System

Regulatory Evaluation

The spent fuel pool provides wet storage of spent fuel assemblies. The safety function of the spent fuel pool cooling and cleanup system is to cool the spent fuel assemblies and keep the spent fuel assemblies covered with water during all storage conditions. The NRC staff's review for the proposed EPU focused on the effects of the proposed EPU on the capability of the system to provide adequate cooling to the spent fuel during all operating and accident conditions. The NRC's acceptance criteria for the spent fuel pool cooling and cleanup system are based on (1) draft GDC-4, insofar as it requires that reactor facilities shall not share systems or components unless it is shown safety is not impaired by the sharing; and (2) draft GDC-67, insofar as it requires that reliable decay heat removal systems be designed to prevent damage to the fuel in storage. Specific review criteria are contained in SRP Section 9.1.3, as supplemented by the guidance provided in Attachment 1 to Matrix 5 of Section 2.1 of RS-001.

Technical Evaluation

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

Conclusion

The NRC staff has reviewed the licensee's assessment related to the spent fuel pool cooling and cleanup system and concludes that the licensee has adequately accounted for the effects of the proposed EPU on the spent fuel pool cooling function of the system. Based on this review, the NRC staff concludes that the spent fuel pool cooling and cleanup system will continue to provide sufficient cooling capability to cool the spent fuel pool following implementation of the proposed EPU and will continue to meet the requirements of draft GDC-4 and 67. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the spent fuel pool cooling and cleanup system.

2.5.3.2 Station Service Water System

Regulatory Evaluation

The station service water system (SWS) provides essential cooling to safety-related equipment and may also provide cooling to non-safety-related auxiliary components that are used for normal plant operation. The NRC staff's review covered the characteristics of the station SWS components with respect to their functional performance as affected by adverse operational (i.e., water hammer) conditions, abnormal operational conditions, and accident conditions (e.g., a LOCA with the LOOP). The NRC staff's review focused on the additional heat load that would result from the proposed EPU. The NRC's acceptance criteria are based on (1) draft GDC-40 and 42, insofar as they require that protection be provided for ESFs against the dynamic effects that might result from plant equipment failures, as well as the effects of a loss of coolant accident; and (2) draft GDC-4, insofar as it requires that reactor facilities shall not share systems or components unless it is shown safety is not impaired by the sharing. Specific review criteria are contained in SRP Section 9.2.1, as supplemented by GL 89-13 and GL 96-06.

Technical Evaluation

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

Conclusion

The NRC staff has reviewed the licensee's assessment related to the effects of the proposed EPU on the station SWS and concludes that the licensee has adequately accounted for the increased heat loads on system performance that would result from the proposed EPU. The NRC staff concludes that the station SWS will continue to be protected from the dynamic effects associated with flow instabilities and provide sufficient cooling for SSCs important to safety following implementation of the proposed EPU. Therefore, the NRC staff has determined that the station SWS will continue to meet the requirements of draft GDC-4, 40 and 42. Based on the above, the NRC staff finds the proposed EPU acceptable with respect to the station SWS.

2.5.3.3 Reactor Auxiliary Cooling Water Systems

[Not applicable. BFN does not have a Reactor Auxiliary Cooling Water System.]

Regulatory Evaluation

Technical Evaluation

Conclusion

2.5.3.4 Ultimate Heat Sink

Regulatory Evaluation

The ultimate heat sink (UHS) is the source of cooling water provided to dissipate reactor decay heat and essential cooling system heat loads after a normal reactor shutdown or a shutdown following an accident. The NRC staff's review focused on the impact that the proposed EPU has on the decay heat removal capability of the UHS. Additionally, the NRC staff's review included evaluation of the design-basis UHS temperature limit determination to confirm that post-licensing data trends (e.g., air and water temperatures, humidity, wind speed, water volume) do not establish more severe conditions than previously assumed. The NRC's acceptance criteria for the UHS are based on draft GDC-4, insofar as it requires that reactor facilities shall not share systems or components unless it is shown safety is not impaired by the sharing. Specific review criteria are contained in SRP Section 9.2.5.

Technical Evaluation

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

Conclusion

The NRC staff has reviewed the information that was provided by the licensee for addressing the effects that the proposed EPU would have on the UHS safety function, including the licensee's validation of the design-basis UHS temperature limit based on post-licensing data. Based on the information that was provided, the NRC staff concludes that the proposed EPU will not compromise the design-basis safety function of the UHS, and that the UHS will continue to satisfy the requirements of draft GDC 4 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the UHS.

2.5.4 Balance-of-Plant Systems

2.5.4.1. Main Steam

Regulatory Evaluation

The main steam supply system (MSSS) transports steam from the NSSS to the power conversion system and various safety-related and non-safety-related auxiliaries. The NRC staff's review focused on the effects of the proposed EPU on the system's capability to transport steam to the power conversion system, provide heat sink capacity, supply steam to drive safety system pumps, and withstand adverse dynamic loads (e.g., water steam hammer resulting from rapid valve closure and relief valve fluid discharge loads). The NRC's acceptance criteria for the MSSS are based on (1) draft GDC-40 and 42, insofar as they require that protection be provided for ESFs against the dynamic effects and missiles that might result from plant equipment failures, as well as the effects of a loss of coolant accident; and (2) draft GDC-4, insofar as it requires that reactor facilities shall not share systems or components unless it is shown safety is not impaired by the sharing. Specific review criteria are contained in SRP Section 10.3.

Technical Evaluation

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

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the MSSS and concludes that the licensee has adequately accounted for the effects of changes in plant conditions on the design of the MSSS. The NRC staff concludes that the MSSS will maintain its ability to transport steam to the power conversion system, provide heat sink capacity, supply steam to steam-driven safety pumps, and withstand steam hammer. The NRC staff further concludes that the MSSS will continue to meet the requirements of draft GDC-4, 40 and 42. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the MSSS.

2.5.4.2 Main Condenser

Regulatory Evaluation

The main condenser (MC) system is designed to condense and deaerate the exhaust steam from the main turbine and provide a heat sink for the turbine bypass system (TBS). For BWRs without an MSIV leakage control system, the MC system may also serve an accident mitigation function to act as a holdup volume for the plateout of fission products leaking through the MSIVs following core damage. The NRC staff's review focused on the effects of the proposed EPU on the steam bypass capability with respect to load rejection assumptions, and on the ability of the MC system to withstand the blowdown effects of steam from the TBS. The NRC's acceptance criteria for the MC system are based on draft GDC-70, insofar as it requires that the plant design include means to control the release of radioactive effluents. Specific review criteria are contained in SRP Section 10.4.1.

Technical Evaluation

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

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the MC system and concludes that the licensee has adequately accounted for the effects of changes in plant conditions on the design of the MC system. The NRC staff concludes that the MC system will continue to maintain its ability to withstand the blowdown effects of the steam from the TBS and thereby continue to meet draft GDC-70 with respect to controlling releases of radioactive effluents. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the MC system.

2.5.4.3 Turbine Bypass

Regulatory Evaluation

The TBS is designed to discharge a stated percentage of rated main steam flow directly to the MC system, bypassing the turbine. This steam bypass enables the plant to take step-load reductions up to the TBS capacity without the reactor or turbine tripping. The system is also used during startup and shutdown to control reactor pressure. For a BWR without an MSIV leakage control system, the TBS could also provide an accident mitigation function. A TBS, along with the MSSS and MC system, may be credited for mitigating the effects of MSIV leakage during a LOCA by the holdup and plateout of fission products. The NRC staff's review for the TBS focused on the effects that the proposed EPU have on load rejection capability, analysis of postulated system piping failures, and the consequences of inadvertent TBS operation. The NRC's acceptance criteria for the TBS are based on draft GDC-40 and 42, insofar as they require that protection be provided for ESFs against the dynamic effects that might result from plant equipment failures, as well as the effects of a loss of coolant accident. Specific review criteria are contained in SRP Section 10.4.4.

Technical Evaluation

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

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the TBS. The NRC staff concludes that the licensee has adequately accounted for the effects of changes in plant conditions on the design of the TBS. The NRC staff concludes that the TBS will continue to mitigate the effects of MSIV leakage during a LOCA and provide a means for shutting down the plant during normal operations. The NRC staff further concludes that TBS failures will not adversely affect essential SSCs. Based on this, the NRC staff concludes that the TBS will continue to meet draft GDC-40 and 42. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the TBS.

2.5.4.4 Condensate and Feedwater

Regulatory Evaluation

The condensate and feedwater system (CFS) provides feedwater at a particular temperature, pressure, and flow rate to the reactor. The only part of the CFS classified as safety-related is the feedwater piping from the NSSS up to and including the outermost containment isolation valve. The NRC staff's review focused on how the proposed EPU affects previous analyses and considerations with respect to the capability of the CFS to supply adequate feedwater during plant operation and shutdown, and isolate components, subsystems, and piping in order to preserve the system's safety function. The NRC's acceptance criteria for the CFS are based on (1) draft GDC-40 and 42, insofar as they require that protection be provided for ESFs against the dynamic effects that might result from plant equipment failures, as well as the effects of a loss of coolant accident; and (2) draft GDC-4, insofar as it requires that reactor facilities shall not share systems or components unless it is shown safety is not impaired by the sharing. Specific review criteria are contained in SRP Section 10.4.7.

Technical Evaluation

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

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the CFS and concludes that the licensee has adequately accounted for the effects of changes in plant conditions on the design of the CFS. The NRC staff concludes that the CFS will continue to maintain its ability to satisfy feedwater requirements for normal operation and shutdown, withstand water hammer, maintain isolation capability in order to preserve the system safety function, and not cause failure of safety-related SSCs. The NRC staff further concludes that the CFS will continue to meet the requirements of draft GDC-4, 40, and 42. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the CFS.

2.5.5 Waste Management Systems

2.5.5.1 Gaseous Waste Management Systems

Regulatory Evaluation

The gaseous waste management systems involve the gaseous radwaste system, which deals with the management of radioactive gases collected in the offgas system or the waste gas storage and decay tanks. In addition, it involves the management of the condenser air removal system; the gland seal exhaust and the mechanical vacuum pump operation exhaust; and the building ventilation system exhausts. The NRC staff's review focused on the effects that the proposed EPU may have on (1) the design criteria of the gaseous waste management systems, (2) methods of treatment, (3) expected releases, (4) principal parameters used in calculating the releases of radioactive materials in gaseous effluents, and (5) design features for precluding the possibility of an explosion if the potential for explosive mixtures exists. The NRC's acceptance criteria for gaseous waste management systems are based on (1) 10 CFR 20.1302, insofar as it provides for demonstrating that annual average concentrations of radioactive materials released at the boundary of the unrestricted area do not exceed specified values; (2) draft GDC-3, insofar as it requires that the reactor facility shall be designed (1) to minimize the probability of events, such as fire and explosions and (2) to minimize the potential effects of such events to safety; (3) draft GDC-70, insofar as it requires that the plant design include means to control the release of radioactive effluents; (4) draft GDC-67, 68, and 69, insofar as they require that systems that contain radioactivity be designed with appropriate confinement; and (5) 10 CFR Part 50, Appendix I, Sections II.B, II.C, and II.D, which set numerical guides for design objectives and limiting conditions for operation to meet the "as low as is reasonably achievable" (ALARA) criterion. Specific review criteria are contained in SRP Section 11.3.

Technical Evaluation

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

Conclusion

The NRC staff has reviewed the licensee's assessment related to the gaseous waste management systems. The NRC staff concludes that the licensee has adequately accounted for the effects of the increase in fission product and amount of gaseous waste on the abilities of the systems to control releases of radioactive materials and preclude the possibility of an explosion if the potential for explosive mixtures exists. The NRC staff finds that the gaseous waste management systems will continue to meet their design functions following implementation of the proposed EPU. The NRC staff further concludes that the licensee has demonstrated that the gaseous waste management systems will continue to meet the requirements of 10 CFR 20.1302; draft GDC-3, 67, 68, 69, and 70; and 10 CFR Part 50, Appendix I, Sections II.B, II.C, and II.D. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the gaseous waste management systems.

2.5.5.2 Liquid Waste Management Systems

Regulatory Evaluation

The NRC staff's review for liquid waste management systems focused on the effects that the proposed EPU may have on previous analyses and considerations related to the liquid waste management systems' design, design objectives, design criteria, methods of treatment, expected releases, and principal parameters used in calculating the releases of radioactive materials in liquid effluents. The NRC's acceptance criteria for the liquid waste management systems are based on (1) 10 CFR 20.1302, insofar as it provides for demonstrating that annual average concentrations of radioactive materials released at the boundary of the unrestricted area do not exceed specified values; (2) draft GDC-70, insofar as it requires that the plant design include means to control the release of radioactive effluents; (3) draft GDC-67, 68, and 69, insofar as they require that systems that contain radioactivity be designed with appropriate confinement; and (4) 10 CFR Part 50, Appendix I, Sections II.A and II.D, which set numerical guides for dose design objectives and limiting conditions for operation to meet the ALARA criterion. Specific review criteria are contained in SRP Section 11.2.

Technical Evaluation

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

Conclusion

The NRC staff has reviewed the licensee's assessment related to the liquid waste management systems. The NRC staff concludes that the licensee has adequately accounted for the effects of the increase in fission product and amount of liquid waste on the ability of the liquid waste management systems to control releases of radioactive materials. The NRC staff finds that the liquid waste management systems will continue to meet their design functions following implementation of the proposed EPU. The NRC staff further concludes that the licensee has demonstrated that the liquid waste management systems will continue to meet the requirements of 10 CFR 20.1302; draft GDC-67, 68, 69, and 70; and 10 CFR Part 50, Appendix I, Sections II.A and II.D. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the liquid waste management systems.

2.5.5.3 Solid Waste Management Systems

Regulatory Evaluation

The NRC staff's review for the solid waste management systems (SWMS) focused on the effects that the proposed EPU may have on previous analyses and considerations related to the design objectives in terms of expected volumes of waste to be processed and handled, the wet and dry types of waste to be processed, the activity and expected radionuclide distribution contained in the waste, equipment design capacities, and the principal parameters employed in the design of the SWMS. The NRC's acceptance criteria for the SWMS are based on (1) 10 CFR 20.1302, insofar as it provides for demonstrating that annual average concentrations of radioactive materials released at the boundary of the unrestricted area do not exceed specified values; (2) draft GDC-70, insofar as it requires that the plant design include means to control the release of radioactive effluents; (3) draft GDC-18, insofar as it requires that systems be provided in waste handling areas to detect conditions that may result in excessive radiation levels, (4) draft GDC-17, insofar as it requires that means be provided for monitoring effluent discharge paths and the plant environs for radioactivity that may be released from normal operations, including AOOs, and postulated accidents; and (5) 10 CFR Part 71, which states requirements for radioactive material packaging. Specific review criteria are contained in SRP Section 11.4.

Technical Evaluation

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

Conclusion

The NRC staff has reviewed the licensee's assessment related to the SWMS. The NRC staff concludes that the licensee has adequately accounted for the effects of the increase in fission product and amount of solid waste on the ability of the SWMS to process the waste. The NRC staff finds that the SWMS will continue to meet its design functions following implementation of the proposed EPU. The NRC staff further concludes that the licensee has demonstrated that the SWMS will continue to meet the requirements of 10 CFR 20.1302, draft GDC-17, 18, and 70, and 10 CFR Part 71. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the SWMS.

2.5.6 Additional Considerations

2.5.6.1 Emergency Diesel Engine Fuel Oil Storage and Transfer System

Regulatory Evaluation

Nuclear power plants are required to have redundant onsite emergency power supplies of sufficient capacity to perform their safety functions (e.g., power diesel engine-driven generator sets), assuming a single failure. The NRC staff's review focused on increases in emergency diesel generator electrical demand and the resulting increase in the amount of fuel oil necessary for the system to perform its safety function. The NRC's acceptance criteria for the emergency diesel engine fuel oil storage and transfer system are based on (1) draft GDC-40 and 42, insofar as they require that protection be provided for ESFs against the dynamic effects, including missiles associated with pipe breaks, as well as the effects of a loss of coolant accident; (2) draft GDC-4, insofar as it requires that reactor facilities shall not share systems or components unless it is shown safety is not impaired by the sharing; and (3) draft GDC-39, insofar as it requires onsite power supplies to have sufficient independence and redundancy to perform their safety functions, assuming a single failure. Specific review criteria are contained in SRP Section 9.5.4.

Technical Evaluation

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

Conclusion

The NRC staff has reviewed the licensee's assessment related to the amount of required fuel oil for the emergency diesel generators and concludes that the licensee has adequately accounted for the effects of the increased electrical demand on fuel oil consumption. The NRC staff concludes that the fuel oil storage and transfer system will continue to provide an adequate amount of fuel oil to allow the diesel generators to meet the onsite power requirements of draft GDC-4, 39, 40, and 42. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the fuel oil storage and transfer system.

2.5.6.2 Light Load Handling System (Related to Refueling)

Regulatory Evaluation

The light load handling system (LLHS) includes components and equipment used in handling new fuel at the receiving station and the loading of spent fuel into shipping casks. The NRC staff's review covered the avoidance of criticality accidents, radioactivity releases resulting from damage to irradiated fuel, and unacceptable personnel radiation exposures. The NRC staff's review focused on the effects of the new fuel on system performance and related analyses. The NRC's acceptance criteria for the LLHS are based on (1) draft GDC-67, 68, and 69, insofar as they require that systems that contain radioactivity be designed with appropriate confinement and with suitable shielding for radiation protection; and (2) draft GDC-66, insofar as it requires that criticality be prevented. Specific review criteria are contained in SRP Section 9.1.4.

Technical Evaluation

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

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the new fuel on the ability of the LLHS to avoid criticality accidents and concludes that the licensee has adequately incorporated the effects of the new fuel in the analyses. Based on this review, the NRC staff further concludes that the LLHS will continue to meet the requirements of draft GDC-66, 67, 68, and 69 for radioactivity releases and prevention of criticality accidents. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the LLHS.

[2.5.7 Additional Review Areas (Plant Systems)]

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

INSERT 6

FOR

SECTION 3.2 - BWR TEMPLATE SAFETY EVALUATION

2.6 Containment Review Considerations

2.6.1 Primary Containment Functional Design

Regulatory Evaluation

The containment encloses the reactor system and is the final barrier against the release of significant amounts of radioactive fission products in the event of an accident. The NRC staff's review for the primary containment functional design covered (1) the temperature and pressure conditions in the drywell and wetwell due to a spectrum of postulated LOCAs, (2) the differential pressure across the operating deck for a spectrum of LOCAs (Mark II containments only), (3) suppression pool dynamic effects during a LOCA or following the actuation of one or more RCS safety/relief valves, (4) the consequences of a LOCA occurring within the containment (wetwell), (5) the capability of the containment to withstand the effects of steam bypassing the suppression pool, (6) the suppression pool temperature limit during RCS safety/relief valve operation, and (7) the analytical models used for containment analysis. The NRC's acceptance criteria for the primary containment functional design are based on (1) draft GDC-40 and 42, insofar as they require that protection be provided for ESFs against the dynamic effects that might result from plant equipment failures, as well as the effects of a loss of coolant accident; (2) draft GDC-10, insofar as it requires that reactor containment be designed to sustain the initial effects of gross equipment failures, such as a large coolant boundary break, without loss of required integrity and, together with other engineered safety features as may be necessary, to retain for as long as the situation requires the functional capability; (3) draft GDC-49, insofar as it requires that the containment and its associated heat removal systems be designed so that the containment structure can accommodate, without exceeding the design leakage rate the pressures and temperatures resulting from the largest credible energy release following a loss-of-coolant accident, including considerable margin for effects from metal-water or other chemical reactions that could occur as a consequence of failure of emergency core cooling systems; (4) draft GDC-12, insofar as it requires that instrumentation and controls be provided as required to monitor and maintain variables within prescribed operating ranges; and (5) draft GDC-17, insofar as it requires that means be provided to monitor the reactor containment atmosphere for radioactivity that may be released from normal operations and from postulated accidents. Specific review criteria are contained in SRP Section 6.2.1.1.C.

Technical Evaluation

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

Conclusion

The NRC staff has reviewed the licensee's assessment of the containment temperature and pressure transient and concludes that the licensee has adequately accounted for the increase of mass and energy resulting from the proposed EPU. The NRC staff further concludes that containment systems will continue to provide sufficient pressure and temperature mitigation capability to ensure that containment integrity is maintained. The NRC staff also concludes that containment systems and instrumentation will continue to be adequate for monitoring containment parameters and release of radioactivity during normal and accident conditions and the containment and associated systems will continue to meet the requirements of draft GDC-10, 12, 17, 40, 42, and 49 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to primary containment functional design.

2.6.2 Subcompartment Analyses

Regulatory Evaluation

A subcompartment is defined as any fully or partially enclosed volume within the primary containment that houses high-energy piping and would limit the flow of fluid to the main containment volume in the event of a postulated pipe rupture within the volume. The NRC staff's review for subcompartment analyses covered the determination of the design differential pressure values for containment subcompartments. The NRC staff's review focused on the effects of the increase in mass and energy release into the containment due to operation at EPU conditions, and the resulting increase in pressurization. The NRC's acceptance criteria for subcompartment analyses are based on (1) draft GDC-40 and 42, insofar as they require that protection be provided for ESFs against the dynamic effects that might result from plant equipment failures, as well as the effects of a loss of coolant accident; and (2) draft GDC-49, insofar as it requires that the containment structure, including access openings and penetrations, and any necessary containment heat removal systems be designed so that the containment structure can accommodate without exceeding the design leakage rate the pressures and temperatures resulting from the largest credible energy release following a loss-of-coolant accident. Specific review criteria are contained in SRP Section 6.2.1.2.

Technical Evaluation

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

Conclusion

The NRC staff has reviewed the subcompartment assessment performed by the licensee and the change in predicted pressurization resulting from the increased mass and energy release. The NRC staff concludes that containment SSCs important to safety will continue to be protected from the dynamic effects resulting from pipe breaks and that the subcompartments will continue to have sufficient margins to prevent fracture of the structure due to pressure difference across the walls following implementation of the proposed EPU. Based on this, the NRC staff concludes that the plant will continue to meet draft GDC-40, 42, and 49 for the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to subcompartment analyses.

2.6.3 Mass and Energy Release

2.6.3.1 Mass and Energy Release Analysis for Postulated Loss of Coolant

Regulatory Evaluation

The release of high-energy fluid into containment from pipe breaks could challenge the structural integrity of the containment, including subcompartments and systems within the containment. The NRC staff's review covered the energy sources that are available for release to the containment and the mass and energy release rate calculations for the initial blowdown phase of the accident. The NRC's acceptance criteria for mass and energy release analyses for postulated LOCAs are based on (1) draft GDC-49, insofar as it requires that the containment structure be designed to accommodate, without exceeding the design leakage rate, the pressures and temperatures resulting from the largest credible energy release following a LOCA; and (2) 10 CFR Part 50, Appendix K, insofar as it identifies sources of energy during a LOCA. Specific review criteria are contained in SRP Section 6.2.1.3.

Technical Evaluation

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

Conclusion

The NRC staff has reviewed the licensee's mass and energy release assessment and concludes that the licensee has adequately addressed the effects of the proposed EPU and appropriately accounts for the sources of energy identified in 10 CFR Part 50, Appendix K. Based on this, the NRC staff finds that the mass and energy release analysis meets the requirements in draft GDC-49 for ensuring that the analysis is conservative. Therefore, the NRC staff finds the proposed EPU acceptable with respect to mass and energy release for postulated LOCA.

2.6.4 Combustible Gas Control in Containment

Regulatory Evaluation

Following a LOCA, hydrogen and oxygen may accumulate inside the containment due to chemical reactions between the fuel rod cladding and steam, corrosion of aluminum and other materials, and radiolytic decomposition of water. If excessive hydrogen is generated, it may form a combustible mixture in the containment atmosphere. The NRC staff's review covered (1) the production and accumulation of combustible gases, (2) the capability to prevent high concentrations of combustible gases in local areas, (3) the capability to monitor combustible gas concentrations, and (4) the capability to reduce combustible gas concentrations. The NRC staff's review primarily focused on any impact that the proposed EPU may have on hydrogen release assumptions, and how increases in hydrogen release are mitigated. The NRC's acceptance criteria for combustible gas control in containment are based on (1) 10 CFR 50.44, insofar as it requires that plants be provided with the capability for controlling combustible gas concentrations in the containment atmosphere; (2) draft GDC-4, insofar as it requires that reactor facilities shall not share systems or components unless it is shown safety is not impaired by the sharing; (3) draft GDC-62, insofar as it requires that all critical parts of containment air cleanup systems, such as ducts, filters, fans, and dampers be designed to permit physical inspection; and (4) draft GDC-63, 64, and 65, insofar as they require that active components of the air cleanup systems be designed to permit appropriate periodic testing. **[*Include the following sentence for BWRs with Mark III containments:* Additional requirements based on 10 CFR 50.44 for control of combustible gas apply to plants with a Mark III type of containment that do not rely on an inerted atmosphere to control hydrogen inside the containment.]** Specific review criteria are contained in SRP Section 6.2.5.

Technical Evaluation

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

Conclusion

The NRC staff has reviewed the licensee's assessment related to combustible gas and concludes that the plant will continue to have sufficient capabilities consistent with the requirements in 10 CFR 50.44 and draft GDC-4, 62, 63, 64, and 65 as discussed above. Therefore, the NRC staff finds the proposed EPU acceptable with respect to combustible gas control in containment.

2.6.5 Containment Heat Removal

Regulatory Evaluation

Fan cooler systems, spray systems, and residual heat removal (RHR) systems are provided to remove heat from the containment atmosphere and from the water in the containment wetwell. The NRC staff's review in this area focused on (1) the effects of the proposed EPU on the analyses of the available net positive suction head (NPSH) to the containment heat removal system pumps and (2) the analyses of the heat removal capabilities of the spray water system and the fan cooler heat exchangers. The NRC's acceptance criteria for containment heat removal are based on draft GDC-41 and 52, insofar as they require that a containment heat removal system be provided, and that its function shall be to prevent exceeding containment design pressure under accident conditions. Specific review criteria are contained in SRP Section 6.2.2, as supplemented by Draft Guide (DG) 1107.

Technical Evaluation

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

Conclusion

The NRC staff has reviewed the containment heat removal systems assessment provided by the licensee and concludes that the licensee has adequately addressed the effects of the proposed EPU. The NRC staff finds that the systems will continue to meet draft GDC-41 and 52 with respect to limiting the containment pressure and temperature following a LOCA and maintaining them at acceptably low levels. Therefore, the NRC staff finds the proposed EPU acceptable with respect to containment heat removal systems.

2.6.6 Secondary Containment Functional Design

Regulatory Evaluation

The secondary containment structure and supporting systems of dual containment plants are provided to collect and process radioactive material that may leak from the primary containment following an accident. The supporting systems maintain a negative pressure within the secondary containment and process this leakage. The NRC staff's review covered (1) analyses of the pressure and temperature response of the secondary containment following accidents within the primary and secondary containments; (2) analyses of the effects of openings in the secondary containment on the capability of the depressurization and filtration system to establish a negative pressure in a prescribed time; (3) analyses of any primary containment leakage paths that bypass the secondary containment; (4) analyses of the pressure response of the secondary containment resulting from inadvertent depressurization of the primary containment when there is vacuum relief from the secondary containment; and (5) the acceptability of the mass and energy release data used in the analysis. The NRC staff's review primarily focused on the effects that the proposed EPU may have on the pressure and temperature response and drawdown time of the secondary containment, and the impact this may have on offsite dose. The NRC's acceptance criteria for secondary containment functional design are based on (1) draft GDC-40 and 42, insofar as they require that protection be provided for ESFs against the dynamic effects that might result from plant equipment failures, as well as the effects of a loss of coolant accident; and (2) draft GDC-10, insofar as it requires that reactor containment be designed to sustain the initial effects of gross equipment failures, such as a large coolant boundary break, without loss of required integrity and, together with other engineered safety features as may be necessary, to retain functional capability for as long as the situation requires. Specific review criteria are contained in SRP Section 6.2.3.

Technical Evaluation

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

Conclusion

The NRC staff has reviewed the licensee's assessment related to the secondary containment pressure and temperature transient and the ability of the secondary containment to provide an essentially leak-tight barrier against uncontrolled release of radioactivity to the environment. The NRC staff concludes that the licensee has adequately accounted for the increase of mass and energy that would result from the proposed EPU and further concludes that the secondary containment and associated systems will continue to provide an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment following implementation of the proposed EPU. Based on this, the NRC staff also concludes that the secondary containment and associated systems will continue to meet the requirements of draft GDC-10, 40, and 42. Therefore, the NRC staff finds the proposed EPU acceptable with respect to secondary containment functional design.

[2.6.7 Additional Review Areas (Containment Review Considerations)]

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

INSERT 7

FOR

SECTION 3.2 - BWR TEMPLATE SAFETY EVALUATION

2.7 Habitability, Filtration, and Ventilation

2.7.1 Control Room Habitability System

Regulatory Evaluation

The NRC staff reviewed the control room habitability system and control building layout and structures to ensure that plant operators are adequately protected from the effects of accidental releases of toxic and radioactive gases. A further objective of the NRC staff's review was to ensure that the control room can be maintained as the backup center from which technical support center personnel can safely operate in the case of an accident. The NRC staff's review focused on the effects of the proposed EPU on radiation doses, toxic gas concentrations, and estimates of dispersion of airborne contamination. The NRC's acceptance criteria for the control room habitability system are based on (1) draft GDC-40 and 42, insofar as they require that protection be provided for ESFs against the dynamic effects that might result from plant equipment failures, as well as the effects of a loss of coolant accident; and (2) draft GDC-11 and 10CFR50.67, insofar as they require that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem TEDE for the duration of the accident. Specific review criteria are contained in SRP Section 6.4 and other guidance provided in Matrix 7 of RS-001.

Technical Evaluation

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

Conclusion

The NRC staff has reviewed the licensee's assessment related to the effects of the proposed EPU on the ability of the control room habitability system to protect plant operators against the effects of accidental releases of toxic and radioactive gases. The NRC staff concludes that the licensee has adequately accounted for the increase of toxic and radioactive gases that would result from the proposed EPU. The NRC staff further concludes that the control room habitability system will continue to provide the required protection following implementation of the proposed EPU. Based on this, the NRC staff concludes that the control room habitability system will continue to meet the requirements of draft GDC-11, 40, and 42, and 10CFR50.67. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the control room habitability system.

2.7.2 Engineered Safety Feature Atmosphere Cleanup

Regulatory Evaluation

ESF atmosphere cleanup systems are designed for fission product removal in postaccident environments. These systems generally include primary systems (e.g., in-containment recirculation) and secondary systems (e.g., standby gas treatment systems and emergency or postaccident air-cleaning systems) for the fuel-handling building, control room, shield building, and areas containing ESF components. For each ESF atmosphere cleanup system, the NRC staff's review focused on the effects of the proposed EPU on system functional design, environmental design, and provisions to preclude temperatures in the adsorber section from exceeding design limits. The NRC's acceptance criteria for ESF atmosphere cleanup systems are based on (1) draft GDC-11 and 10CFR50.67, insofar as they require that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem TEDE for the duration of the accident; (2) draft GDC-67, 68, and 69, insofar as they require that systems that may contain radioactivity be designed to assure adequate safety under normal and postulated accident conditions; and (4) draft GDC-17, insofar as it requires that means be provided for monitoring effluent discharge paths and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences (AOOs), and postulated accidents. Specific review criteria are contained in SRP Section 6.5.1.

Technical Evaluation

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

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the ESF atmosphere cleanup systems. The NRC staff concludes that the licensee has adequately accounted for the increase of fission products and changes in expected environmental conditions that would result from the proposed EPU, and the NRC staff further concludes that the ESF atmosphere cleanup systems will continue to provide adequate fission product removal in postaccident environments following implementation of the proposed EPU. Based on this, the NRC staff concludes that the ESF atmosphere cleanup systems will continue to meet the requirements of draft GDC-11, 17, 67, 68, and 69; and 10CFR50.67. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the ESF atmosphere cleanup systems.

2.7.3 Control Room Area Ventilation System

Regulatory Evaluation

The function of the control room area ventilation system (CRAVS) is to provide a controlled environment for the comfort and safety of control room personnel and to support the operability of control room components during normal operation, AOOs, and DBA conditions. The NRC's review of the CRAVS focused on the effects that the proposed EPU will have on the functional performance of safety-related portions of the system. The review included the effects of radiation, combustion, and other toxic products; and the expected environmental conditions in areas served by the CRAVS. The NRC's acceptance criteria for the CRAVS are based on (1) draft GDC-40 and 42, insofar as they require that protection be provided for ESFs against the dynamic effects that might result from plant equipment failures, as well as the effects of a loss of coolant accident; (2) draft GDC-11 and 10CFR50.67, insofar as they require that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem TEDE for the duration of the accident; and (3) draft GDC-70, insofar as it requires that the plant design include means to control the release of radioactive effluents. Specific review criteria are contained in SRP Section 9.4.1.

Technical Evaluation

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

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the ability of the CRAVS to provide a controlled environment for the comfort and safety of control room personnel and to support the operability of control room components. The NRC staff concludes that the licensee has adequately accounted for the increase of toxic and radioactive gases that would result from a DBA under the conditions of the proposed EPU, and associated changes to parameters affecting environmental conditions for control room personnel and equipment. Accordingly, the NRC staff concludes that the CRAVS will continue to provide an acceptable control room environment for safe operation of the plant following implementation of the proposed EPU. The NRC staff also concludes that the system will continue to suitably control the release of gaseous radioactive effluents to the environment. Based on this, the NRC staff concludes that the CRAVS will continue to meet the requirements of draft GDC-11, 40, 42, and 70, and 10CFR50.67. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the CRAVS.

2.7.4 Spent Fuel Pool Area Ventilation System

[Not applicable. BFN does not have a separate Spent Fuel Pool Area Ventilation System.]

Regulatory Evaluation

Technical Evaluation

Conclusion

2.7.5 Auxiliary and Radwaste Area and Turbine Areas Ventilation Systems

Regulatory Evaluation

The function of the auxiliary and radwaste area ventilation system (ARAVS) and the turbine area ventilation system (TAVS) is to maintain ventilation in the auxiliary and radwaste equipment and turbine areas, permit personnel access, and control the concentration of airborne radioactive material in these areas during normal operation, during AOOs, and after postulated accidents. The NRC staff's review focused on the effects of the proposed EPU on the functional performance of the safety-related portions of these systems. The NRC's acceptance criteria for the ARAVS and TAVS are based on draft GDC-70, insofar as it requires that the plant design include means to control the release of radioactive effluents. Specific review criteria are contained in SRP Sections 9.4.3 and 9.4.4.

Technical Evaluation

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

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the ARAVS and TAVS. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed EPU on the capability of these systems to maintain ventilation in the auxiliary and radwaste equipment areas and in the turbine area, permit personnel access, control the concentration of airborne radioactive material in these areas, and control release of gaseous radioactive effluents to the environment. Based on this, the NRC staff concludes that the ARAVS and TAVS will continue to meet the requirements of draft GDC-70. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the ARAVS and the TAVS.

2.7.6 Engineered Safety Feature Ventilation System

Regulatory Evaluation

The function of the engineered safety feature ventilation system (ESFVS) is to provide a suitable and controlled environment for ESF components following certain anticipated transients and DBAs. The NRC staff's review for the ESFVS focused on the effects of the proposed EPU on the functional performance of the safety-related portions of the system. The NRC staff's review also covered (1) the ability of the ESF equipment in the areas being serviced by the ventilation system to function under degraded ESFVS performance; (2) the capability of the ESFVS to circulate sufficient air to prevent accumulation of flammable or explosive gas or fuel-vapor mixtures from components (e.g., storage batteries and stored fuel); and (3) the capability of the ESFVS to control airborne particulate material (dust) accumulation. The NRC's acceptance criteria for the ESFVS are based on (1) draft GDC-40 and 42, insofar as they require that protection be provided for ESFs against the dynamic effects and missiles that might result from plant equipment failures, as well as the effects of a loss of coolant accident; (2) draft GDC-24 and 39, insofar as they require onsite and offsite electric power systems be provided to permit functioning of the ESFs and protection systems; and (3) draft GDC-70, insofar as it requires that the plant design include means to control the release of radioactive effluents. Specific review criteria are contained in SRP Section 9.4.5.

Technical Evaluation

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

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the ESFVS. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed EPU on the ability of the ESFVS to provide a suitable and controlled environment for ESF components. The NRC staff further concludes that the ESFVS will continue to assure a suitable environment for the ESF components following implementation of the proposed EPU. The NRC staff also concludes that the ESFVS will continue to suitably control the release of gaseous radioactive effluents to the environment following implementation of the proposed EPU. Based on this, the NRC staff concludes that the ESFVS will continue to meet the requirements of draft GDC-24, 39, 40, 42, and 70. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the ESFVS.

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

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

INSERT 8

FOR

SECTION 3.2 - BWR TEMPLATE SAFETY EVALUATION

2.8 Reactor Systems

2.8.1 Fuel System Design

Regulatory Evaluation

The fuel system consists of arrays of fuel rods, burnable poison rods, spacer grids and springs, end plates, channel boxes, and reactivity control rods. The NRC staff reviewed the fuel system to ensure that (1) the fuel system is not damaged as a result of normal operation and AOOs, (2) fuel system damage is never so severe as to prevent control rod insertion when it is required, (3) the number of fuel rod failures is not underestimated for postulated accidents, and (4) coolability is always maintained. The NRC staff's review covered fuel system damage mechanisms, limiting values for important parameters, and performance of the fuel system during normal operation, AOOs, and postulated accidents. The NRC's acceptance criteria are based on (1) 10 CFR 50.46, insofar as it establishes standards for the calculation of emergency core cooling system (ECCS) performance and acceptance criteria for that calculated performance; (2) draft GDC-6, insofar as it requires that the reactor core be designed to function throughout its design lifetime, without exceeding acceptable fuel damage limits; and (3) draft GDC-37, 41, and 44, insofar as they require that a system to provide abundant emergency core cooling be provided to prevent fuel damage following a LOCA. Specific review criteria are contained in SRP Section 4.2 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

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

Conclusion

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed EPU on the fuel system design of the fuel assemblies, control systems, and reactor core. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed EPU on the fuel system and demonstrated that (1) the fuel system will not be damaged as a result of normal operation and AOOs, (2) the fuel system damage will never be so severe as to prevent control rod insertion when it is required, (3) the number of fuel rod failures will not be underestimated for postulated accidents, and (4) coolability will always be maintained. Based on this, the NRC staff concludes that the fuel system and associated analyses will continue to meet the requirements of 10 CFR 50.46, draft GDC-6, 37, 41, and 44 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the fuel system design.

2.8.2 Nuclear Design

Regulatory Evaluation

The NRC staff reviewed the nuclear design of the fuel assemblies, control systems, and reactor core to ensure that fuel design limits will not be exceeded during normal operation and anticipated operational transients, and that the effects of postulated reactivity accidents will not cause significant damage to the RCPB or impair the capability to cool the core. The NRC staff's review covered core power distribution, reactivity coefficients, reactivity control requirements and control provisions, control rod patterns and reactivity worths, criticality, burnup, and vessel irradiation. The NRC's acceptance criteria are based on (1) draft GDC-6, insofar as it requires that the reactor core be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits during any condition of normal operation, including the effects of AOOs; (2) draft GDC-8, insofar as it requires that the reactor core be designed so that the overall power coefficient in the power operating range shall not be positive; (3) draft GDC-7, insofar as it requires that the reactor core be designed to ensure that power oscillations, which could cause damage in excess of acceptable fuel damage limits, are not possible or can be readily suppressed; (4) draft GDC-12, insofar as it requires that instrumentation and controls be provided as required to monitor and maintain variables within prescribed operating ranges; (5) draft GDC-14 and 15, insofar as they require that the protection system be designed to initiate the reactivity control systems automatically to prevent or suppress conditions that could result in exceeding acceptable fuel damage limits and to initiate operation of ESFs under accident situations; (6) draft GDC-31, insofar as it requires that the reactivity control systems be capable of sustaining any single malfunction without causing a reactivity transient which could result in exceeding acceptable fuel damage limits; (7) draft GDC-27 and 28, insofar as they require that at least two independent reactivity control systems be provided, with both systems capable of making and holding the core subcritical from any hot standby or hot operating condition sufficiently fast to prevent exceeding acceptable fuel damage limits; (8) draft GDC-29, insofar as it requires that at least one of the reactivity control systems be capable of making the core subcritical under any condition sufficiently fast to prevent exceeding acceptable fuel damage limits; and (9) draft GDC-32, insofar as it requires that limits, which include considerable margin, be placed on the maximum reactivity worth of control rods or elements and on rates at which reactivity can be increased to ensure that the potential effects of a sudden or large change of reactivity cannot (a) rupture the reactor coolant pressure boundary or (b) disrupt the core, its support structures, or other vessel internals sufficiently to impair the effectiveness of emergency core cooling. Specific review criteria are contained in SRP Section 4.3 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

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

Conclusion

The NRC staff has reviewed the licensee's analyses related to the effect of the proposed EPU on the nuclear design of the fuel assemblies, control systems, and reactor core. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed EPU on the nuclear design and has demonstrated that the fuel design limits will not be exceeded during normal or anticipated operational transients, and that the effects of postulated reactivity accidents will not cause significant damage to the RCPB or impair the capability to cool the core. Based on this evaluation and in coordination with the reviews of the fuel system design, thermal and hydraulic design, and transient and accident analyses, the NRC staff concludes that the nuclear design of the fuel assemblies, control systems, and reactor core will continue to meet the applicable requirements of draft GDC-6, 7, 8, 12, 14, 15, 27, 28, 29, 31, and 32. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the nuclear design.

2.8.3 Thermal and Hydraulic Design

Regulatory Evaluation

The NRC staff reviewed the thermal and hydraulic design of the core and the RCS to confirm that the design (1) has been accomplished using acceptable analytical methods, (2) is equivalent to or a justified extrapolation from proven designs, (3) provides acceptable margins of safety from conditions which would lead to fuel damage during normal reactor operation and AOOs, and (4) is not susceptible to thermal-hydraulic instability. The review also covered hydraulic loads on the core and RCS components during normal operation and DBA conditions and core thermal-hydraulic stability under normal operation and anticipated transients without scram (ATWS) events. The NRC's acceptance criteria are based on (1) draft GDC-6, insofar as it requires that the reactor core be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits during any condition of normal operation, including the effects of AOOs; and (2) draft GDC-7, insofar as it requires that the reactor core, together with reliable controls, ensure that power oscillations, which could cause damage in excess of acceptable fuel damage limits, are not possible or can be readily suppressed. Specific review criteria are contained in SRP Section 4.4 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

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

Conclusion

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed EPU on the thermal and hydraulic design of the core and the RCS. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed EPU on the thermal and hydraulic design and demonstrated that the design (1) has been accomplished using acceptable analytical methods, (2) is **[equivalent to or a justified extrapolation from]** proven designs, (3) provides acceptable margins of safety from conditions that would lead to fuel damage during normal reactor operation and AOOs, and (4) is not susceptible to thermal-hydraulic instability. The NRC staff further concludes that the licensee has adequately accounted for the effects of the proposed EPU on the hydraulic loads on the core and RCS components. Based on this, the NRC staff concludes that the thermal and hydraulic design will continue to meet the requirements of draft GDC-6 and 7 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to thermal and hydraulic design.

2.8.4 Emergency Systems

2.8.4.1 Functional Design of Control Rod Drive System

Regulatory Evaluation

The NRC staff's review covered the functional performance of the control rod drive system (CRDS) to confirm that the system can effect a safe shutdown, respond within acceptable limits during AOOs, and prevent or mitigate the consequences of postulated accidents. The review also covered the CRDS cooling system to ensure that it will continue to meet its design requirements. The NRC's acceptance criteria are based on (1) draft GDC-40 and 42, insofar as they require that protection be provided for ESFs against the dynamic effects and missiles that might result from plant equipment failures, as well as the effects of a loss of coolant accident; (2) draft GDC-26, insofar as it requires that the protection system be designed to fail into a safe state; (3) draft GDC-31, insofar as it requires that the reactivity control systems be capable of sustaining any single malfunction without causing a reactivity transient which could result in exceeding acceptable fuel damage limits; (4) draft GDC-27 and 28, insofar as they require that at least two independent reactivity control systems be provided, with both systems capable of making and holding the core subcritical from any hot standby or hot operating condition, sufficiently fast to prevent exceeding acceptable fuel damage limits; (5) draft GDC-29, insofar as it requires that at least one of the reactivity control systems be capable of making the core subcritical under any condition sufficiently fast to prevent exceeding acceptable fuel damage limits; (6) draft GDC-32, insofar as it requires that limits, which include considerable margin, be placed on the maximum reactivity worth of control rods or elements and on rates at which reactivity can be increased to ensure that the potential effects of a sudden or large change of reactivity cannot (a) rupture the reactor coolant pressure boundary or (b) disrupt the core, its support structures, or other vessel internals sufficiently to impair the effectiveness of emergency core cooling; and (7) 10 CFR 50.62(c)(3), insofar as it requires that all BWRs have an alternate rod injection (ARI) system diverse from the reactor trip system, and that the ARI system have redundant scram air header exhaust valves. Specific review criteria are contained in SRP Section 4.6.

Technical Evaluation

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

Conclusion

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed EPU on the functional design of the CRDS. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed EPU on the system and demonstrated that the system's ability to effect a safe shutdown, respond within acceptable limits, and prevent or mitigate the consequences of postulated accidents will be maintained following the implementation of the proposed EPU. The NRC staff further concludes that the licensee has demonstrated that sufficient cooling exists to ensure the system's design bases will continue to be followed upon implementation of the proposed EPU. Based on this, the NRC staff concludes that the fuel system and associated analyses will continue to meet the requirements of draft GDC-26, 27, 28, 29, 31, 32, 40, and 42, and 10 CFR 50.62(c)(3) following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the functional design of the CRDS.

2.8.4.2 Overpressure Protection During Power Operation

Regulatory Evaluation

Overpressure protection for the RCPB during power operation is provided by relief and safety valves and the reactor protection system. The NRC staff's review covered relief and safety valves on the main steamlines and piping from these valves to the suppression pool. The NRC's acceptance criteria are based on (1) draft GDC-9, insofar as it requires that the RCPB be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime; and (2) draft GDC-33, 34, and 35, insofar as they require that the RCPB be designed to assure that it behaves in a nonbrittle manner and that the probability of rapidly propagating type failures is minimized. Specific review criteria are contained in SRP Section 5.2.2.

Technical Evaluation

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

Conclusion

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed EPU on the overpressure protection capability of the plant during power operation. The NRC staff concludes that the licensee has (1) adequately accounted for the effects of the proposed EPU on pressurization events and overpressure protection features and (2) demonstrated that the plant will continue to have sufficient pressure relief capacity to ensure that pressure limits are not exceeded. Based on this, the NRC staff concludes that the overpressure protection features will continue to meet draft GDC-9, 33, 34, and 35 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to overpressure protection during power operation.

2.8.4.3 Reactor Core Isolation Cooling System

Regulatory Evaluation

The reactor core isolation cooling (RCIC) system serves as a standby source of cooling water to provide a limited decay heat removal capability whenever the main feedwater system is isolated from the reactor vessel. In addition, the RCIC system may provide decay heat removal necessary for coping with a station blackout. The water supply for the RCIC system comes from the condensate storage tank, with a secondary supply from the suppression pool. The NRC staff's review covered the effect of the proposed EPU on the functional capability of the system. The NRC's acceptance criteria are based on (1) draft GDC-40 and 42, insofar as they require that protection be provided for ESFs against the dynamic effects that might result from plant equipment failures, as well as the effects of a loss of coolant accident; (2) draft GDC-4, insofar as it requires that reactor facilities shall not share systems or components unless it is shown safety is not impaired by the sharing; (3) draft GDC-37, insofar as it requires that ESFs be provided to back up the safety provided by the core design, the RCPB, and their protective systems; (4) draft GDC-51 and 57, insofar as they require that piping systems penetrating containment be designed with appropriate features as necessary to protect from an accidental rupture outside containment and the capability to periodically test the operability of the isolation valves to determine if valve leakage is within acceptable limits; and (5) 10 CFR 50.63, insofar as it requires that the plant withstand and recover from an SBO of a specified duration. Specific review criteria are contained in SRP Section 5.4.6

Technical Evaluation

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

Conclusion

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed EPU on the ability of the RCIC system to provide decay heat removal following an isolation of main feedwater event and a station blackout event and the ability of the system to provide makeup to the core following a small break in the RCPB. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed EPU on these events and demonstrated that the RCIC system will continue to provide sufficient decay heat removal and makeup for these events following implementation of the proposed EPU. Based on this, the NRC staff concludes that the RCIC system will continue to meet the requirements of draft GDC-4, 37, 40, 42, 51, and 57, and 10 CFR 50.63 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the RCIC system.

2.8.4.4 Residual Heat Removal System

Regulatory Evaluation

The RHR system is used to cool down the RCS following shutdown. The RHR system is typically a low pressure system which takes over the shutdown cooling function when the RCS temperature is reduced. The NRC staff's review covered the effect of the proposed EPU on the functional capability of the RHR system to cool the RCS following shutdown and provide decay heat removal. The NRC's acceptance criteria are based on (1) draft GDC-40 and 42, insofar as they require that ESFs be protected against dynamic effects; and (2) draft GDC-4, insofar as it requires that reactor facilities shall not share systems or components unless it is shown safety is not impaired by the sharing. Specific review criteria are contained in SRP Section 5.4.7 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

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

Conclusion

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed EPU on the RHR system. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed EPU on the system and demonstrated that the RHR system will maintain its ability to cool the RCS following shutdown and provide decay heat removal. Based on this, the NRC staff concludes that the RHR system will continue to meet the requirements of draft GDC-4, 40 and 42 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the RHR system.

2.8.4.5 Standby Liquid Control System

Regulatory Evaluation

The standby liquid control system (SLCS) provides backup capability for reactivity control independent of the control rod system. The SLCS functions by injecting a boron solution into the reactor to effect shutdown. The NRC staff's review covered the effect of the proposed EPU on the functional capability of the system to deliver the required amount of boron solution into the reactor. The NRC's acceptance criteria are based on (1) draft GDC-27 and 28, insofar as they require that at least two independent reactivity control systems, preferably of different design principles, be provided, with both systems capable of making and holding the core subcritical from any hot standby or hot operating condition, sufficiently fast to prevent exceeding acceptable fuel damage limits; (2) draft GDC-29, insofar as it requires that at least one of the reactivity control systems be capable of making the core subcritical under any condition sufficiently fast to prevent exceeding acceptable fuel damage limits; and (3) 10 CFR 50.62(c)(4), insofar as it requires that the SLCS be capable of reliably injecting a borated water solution into the reactor pressure vessel at a boron concentration, boron enrichment, and flow rate that provides a set level of reactivity control. Specific review criteria are contained in SRP Section 9.3.5 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

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

Conclusion

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed EPU on the SLCS and concludes that the licensee has adequately accounted for the effects of the proposed EPU on the system and demonstrated that the system will continue to provide the function of reactivity control independent of the control rod system following implementation of the proposed EPU. Based on this, the NRC staff concludes that the SLCS will continue to meet the requirements of draft GDC-27, 28, and 29, and 10 CFR 50.62(c)(4) following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the SLCS.

2.8.5 Accident and Transient Analyses

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

Regulatory Evaluation

Excessive heat removal causes a decrease in moderator temperature which increases core reactivity and can lead to a power level increase and a decrease in shutdown margin. Any unplanned power level increase may result in fuel damage or excessive reactor system pressure. Reactor protection and safety systems are actuated to mitigate the transient. The NRC staff's review covered (1) postulated initial core and reactor conditions, (2) methods of thermal and hydraulic analyses, (3) the sequence of events, (4) assumed reactions of reactor system components, (5) functional and operational characteristics of the reactor protection system, (6) operator actions, and (7) the results of the transient analyses. The NRC's acceptance criteria are based on (1) draft GDC-6, insofar as it requires that the reactor core be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits; (2) draft GDC-14 and 15, insofar as they require that the core protection system be designed to act automatically to prevent or suppress conditions that could result in exceeding acceptable fuel damage limits and that protection systems be provided for sensing accident situations and initiating the operation of necessary ESFs; and (3) draft GDC-27 and 28, insofar as they require that at least two reactivity control systems be provided and be capable of making and holding the core subcritical from any hot standby or hot operating condition sufficiently fast to prevent exceeding acceptable fuel damage limits. Specific review criteria are contained in SRP Section 15.1.1-4 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

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

Conclusion

The NRC staff has reviewed the licensee's analyses of the excess heat removal events described above and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the AFDLs 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 draft GDC-6, 14, 15, 27, and 28 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the events stated.

2.8.5.2 Decrease in Heat Removal by the Secondary System

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

Regulatory Evaluation

A number of initiating events may result in unplanned decreases in heat removal by the secondary system. These events result in a sudden reduction in steam flow and, consequently, result in pressurization events. Reactor protection and safety systems are actuated to mitigate the transient. The NRC staff's review covered the sequence of events, the analytical models used for analyses, the values of parameters used in the analytical models, and the results of the transient analyses. The NRC's acceptance criteria are based on (1) draft GDC-6, insofar as it requires that the reactor core be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits; and (2) draft GDC-27 and 28, insofar as they require that at least two reactivity control systems be provided and be capable of making and holding the core subcritical from any hot standby or hot operating condition sufficiently fast to prevent exceeding acceptable fuel damage limits. Specific review criteria are contained in SRP Section 15.2.1-5 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

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

Conclusion

The NRC staff has reviewed the licensee's analyses of the decrease in heat removal events described above and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the AFDLs 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 draft GDC-6, 27, and 28 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the events stated.

2.8.5.2.2 Loss of Nonemergency AC Power to the Station Auxiliaries

Regulatory Evaluation

The loss of nonemergency ac power is assumed to result in the loss of all power to the station auxiliaries and the simultaneous tripping of all reactor coolant circulation pumps. This causes a flow coastdown as well as a decrease in heat removal by the secondary system, a turbine trip, an increase in pressure and temperature of the coolant, and a reactor trip. Reactor protection and safety systems are actuated to mitigate the transient. The NRC staff's review covered (1) the sequence of events, (2) the analytical model used for analyses, (3) the values of parameters used in the analytical model, and (4) the results of the transient analyses. The NRC's acceptance criteria are based on (1) draft GDC-6, insofar as it requires that the reactor core be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits; and (2) draft GDC-27 and 28, insofar as they require that at least two reactivity control systems be provided and be capable of making and holding the core subcritical from any hot standby or hot operating condition sufficiently fast to prevent exceeding acceptable fuel damage limits. Specific review criteria are contained in SRP Section 15.2.6 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

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

Conclusion

The NRC staff has reviewed the licensee's analyses of the loss of nonemergency ac power to station auxiliaries event and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the AFDLs 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 draft GDC-6, 27, and 28 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the loss of nonemergency ac power to station auxiliaries event.

2.8.5.2.3 Loss of Normal Feedwater Flow

Regulatory Evaluation

A loss of normal feedwater flow could occur from pump failures, valve malfunctions, or a LOOP. Loss of feedwater flow results in an increase in reactor coolant temperature and pressure which eventually requires a reactor trip to prevent fuel damage. Decay heat must be transferred from fuel following a loss of normal feedwater flow. Reactor protection and safety systems are actuated to provide this function and mitigate other aspects of the transient. The NRC staff's review covered (1) the sequence of events, (2) the analytical model used for analyses, (3) the values of parameters used in the analytical model, and (4) the results of the transient analyses. The NRC's acceptance criteria are based on (1) draft GDC-6, insofar as it requires that the reactor core be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits; and (2) draft GDC-27 and 28, insofar as they require that at least two reactivity control systems be provided and be capable of making and holding the core subcritical from any hot standby or hot operating condition sufficiently fast to prevent exceeding acceptable fuel damage limits. Specific review criteria are contained in SRP Section 15.2.7 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

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

Conclusion

The NRC staff has reviewed the licensee's analyses of the loss of normal feedwater flow event and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the AFDLs 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 draft GDC-6, 27, and 28 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the loss of normal feedwater flow event.

2.8.5.3 Decrease in Reactor Coolant System Flow

2.8.5.3.1 Loss of Forced Reactor Coolant Flow

Regulatory Evaluation

A decrease in reactor coolant flow occurring while the plant is at power could result in a degradation of core heat transfer. An increase in fuel temperature and accompanying fuel damage could then result if AFDLs are exceeded during the transient. Reactor protection and safety systems are actuated to mitigate the transient. The NRC staff's review covered (1) the postulated initial core and reactor conditions, (2) the methods of thermal and hydraulic analyses, (3) the sequence of events, (4) assumed reactions of reactor systems components, (5) the functional and operational characteristics of the reactor protection system, (6) operator actions, and (7) the results of the transient analyses. The NRC's acceptance criteria are based on (1) draft GDC-6, insofar as it requires that the reactor core be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits; and (2) draft GDC-27 and 28, insofar as they require that at least two reactivity control systems be provided and be capable of making and holding the core subcritical from any hot standby or hot operating condition sufficiently fast to prevent exceeding acceptable fuel damage limits. Specific review criteria are contained in SRP Section 15.3.1-2 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

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

Conclusion

The NRC staff has reviewed the licensee's analyses of the decrease in reactor coolant flow event and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the AFDLs 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 draft GDC-6, 27, and 28 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the decrease in reactor coolant flow event.

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

Regulatory Evaluation

The events postulated are an instantaneous seizure of the rotor or break of the shaft of a reactor recirculation pump. Flow through the affected loop is rapidly reduced, leading to a reactor and turbine trip. The sudden decrease in core coolant flow while the reactor is at power results in a degradation of core heat transfer which could result in fuel damage. The initial rate of reduction of coolant flow is greater for the rotor seizure event. However, the shaft break event permits a greater reverse flow through the affected loop later during the transient and, therefore, results in a lower core flow rate at that time. In either case, reactor protection and safety systems are actuated to mitigate the transient. The NRC staff's review covered (1) the postulated initial and long-term core and reactor conditions, (2) the methods of thermal and hydraulic analyses, (3) the sequence of events, (4) the assumed reactions of reactor system components, (5) the functional and operational characteristics of the reactor protection system, (6) operator actions, and (7) the results of the transient analyses. The NRC's acceptance criteria are based on (1) draft GDC-32, insofar as it requires that limits, which include considerable margin, be placed on the maximum reactivity worth of control rods or elements and on rates at which reactivity can be increased to ensure that the potential effects of a sudden or large change of reactivity cannot (a) rupture the reactor coolant pressure boundary or (b) disrupt the core, its support structures, or other vessel internals sufficiently to impair the effectiveness of emergency core cooling; and (2) draft GDC-33, 34, and 35, insofar as they require that the RCPB be designed with margin sufficient to assure that, under specified conditions, it will behave in a nonbrittle manner and the probability of rapidly propagating fractures is minimized. Specific review criteria are contained in SRP Section 15.3.3-4 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

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

Conclusion

The NRC staff has reviewed the licensee's analyses of the sudden decrease in core coolant flow events and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the ability to insert control rods is maintained, the RCPB pressure limits will not be exceeded, the RCPB will behave in a nonbrittle manner, the probability of propagating fracture of the RCPB is minimized, and adequate core cooling will be provided. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of draft GDC-32, 33, 34, and 35 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the sudden decrease in core coolant flow events.

2.8.5.4 Reactivity and Power Distribution Anomalies

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

Regulatory Evaluation

An uncontrolled control rod assembly withdrawal from subcritical or low power startup conditions may be caused by a malfunction of the reactor control or rod control systems. This withdrawal will uncontrollably add positive reactivity to the reactor core, resulting in a power excursion. The NRC staff's review covered (1) the description of the causes of the transient and the transient itself, (2) the initial conditions, (3) the values of reactor parameters used in the analysis, (4) the analytical methods and computer codes used, and (5) the results of the transient analyses. The NRC's acceptance criteria are based on (1) draft GDC-6, insofar as it requires that the reactor core be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits; (2) draft GDC-14 and 15, insofar as they require that the core protection systems be designed to act automatically to prevent or suppress conditions that could result in exceeding acceptable fuel damage limits and that protection systems be provided for sensing accident situations and initiating the operation of necessary ESFs; and (3) draft GDC-31, insofar as it requires that the reactivity control systems be capable of sustaining any single malfunction without causing a reactivity transient which could result in exceeding acceptable fuel damage limits. Specific review criteria are contained in SRP Section 15.4.1 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

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

Conclusion

The NRC staff has reviewed the licensee's analyses of the uncontrolled control rod assembly withdrawal from a subcritical or low power startup condition and concludes that the licensee's analyses have adequately accounted for the changes in core design necessary for operation of the plant at the proposed power level. The NRC staff also concludes that the licensee's analyses were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure the AFDLs are not exceeded. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of draft GDC-6, 14, 15, and 31 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the uncontrolled control rod assembly withdrawal from a subcritical or low power startup condition.

2.8.5.4.2 Uncontrolled Control Rod Assembly Withdrawal at Power

Regulatory Evaluation

An uncontrolled control rod assembly withdrawal at power may be caused by a malfunction of the reactor control or rod control systems. This withdrawal will uncontrollably add positive reactivity to the reactor core, resulting in a power excursion. The NRC staff's review covered (1) the description of the causes of the AOO and the description of the event itself, (2) the initial conditions, (3) the values of reactor parameters used in the analysis, (4) the analytical methods and computer codes used, and (5) the results of the associated analyses. The NRC's acceptance criteria are based on (1) draft GDC-6, insofar as it requires that the reactor core be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits; (2) draft GDC-14 and 15, insofar as they require that the core protection systems be designed to act automatically to prevent or suppress conditions that could result in exceeding acceptable fuel damage limits and that protection systems be provided for sensing accident situations and initiating the operation of necessary ESFs; and (3) draft GDC-31, insofar as it requires that the reactivity control systems be capable of sustaining any single malfunction without causing a reactivity transient which could result in exceeding acceptable fuel damage limits. Specific review criteria are contained in SRP Section 15.4.2 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

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

Conclusion

The NRC staff has reviewed the licensee's analyses of the uncontrolled control rod assembly withdrawal at power event and concludes that the licensee's analyses have adequately accounted for the changes in core design required for operation of the plant at the proposed power level. The NRC staff also concludes that the licensee's analyses were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure the AFDLs are not exceeded. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of draft GDC-6, 14, 15, and 31 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the uncontrolled control rod assembly withdrawal at power.

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

Regulatory Evaluation

A startup of an inactive loop transient may result in either an increased core flow or the introduction of cooler water into the core. This event causes an increase in core reactivity due to decreased moderator temperature and core void fraction. The NRC staff's review covered (1) the sequence of events, (2) the analytical model, (3) the values of parameters used in the analytical model, and (4) the results of the transient analyses. The NRC's acceptance criteria are based on (1) draft GDC-6, insofar as it requires that the reactor core be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits; (2) draft GDC-14 and 15, insofar as they require that the core protection systems be designed to act automatically to prevent or suppress conditions that could result in exceeding acceptable fuel damage limits and that protection systems be provided for sensing accident situations and initiating the operation of necessary ESFs; (3) draft GDC-32, insofar as it requires that limits, which include considerable margin, be placed on the maximum reactivity worth of control rods or elements and on rates at which reactivity can be increased to ensure that the potential effects of a sudden or large change of reactivity cannot (a) rupture the reactor coolant pressure boundary or (b) disrupt the core, its support structures, or other vessel internals sufficiently to impair the effectiveness of emergency core cooling; and (4) draft GDC-27 and 28, insofar as they require that at least two reactivity control systems be provided and be capable of making and holding the core subcritical from any hot standby or hot operating condition sufficiently fast to prevent exceeding acceptable fuel damage limits. Specific review criteria are contained in SRP Section 15.4.4-5 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

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

Conclusion

The NRC staff has reviewed the licensee's analyses of the increase in core flow event and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the AFDLs 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 draft GDC-6, 14, 15, 27, 28, and 32 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the increase in core flow event.

2.8.5.4.4 Spectrum of Rod Drop Accidents

Regulatory Evaluation

The NRC staff evaluated the consequences of a control rod drop accident in the area of reactor physics. The NRC staff's review covered the occurrences that lead to the accident, safety features designed to limit the amount of reactivity available and the rate at which reactivity can be added to the core, the analytical model used for analyses, and the results of the analyses. The NRC's acceptance criteria are based on draft GDC-32, insofar as it requires that limits, which include considerable margin, be placed on the maximum reactivity worth of control rods or elements and on rates at which reactivity can be increased to ensure that the potential effects of a sudden or large change of reactivity cannot (a) rupture the reactor coolant pressure boundary or (b) disrupt the core, its support structures, or other vessel internals sufficiently to impair the effectiveness of emergency core cooling. Specific review criteria are contained in SRP Section 15.4.9 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

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

Conclusion

The NRC staff has reviewed the licensee's analyses of the rod drop accident and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that appropriate reactor protection and safety systems will prevent postulated reactivity accidents that could (1) result in damage to the RCPB greater than limited local yielding, or (2) cause sufficient damage that would significantly impair the capability to cool the core. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of draft GDC-32 following implementation of the EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the rod drop accident.

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

Regulatory Evaluation

Equipment malfunctions, operator errors, and abnormal occurrences could cause unplanned increases in reactor coolant inventory. Depending on the temperature of the injected water and the response of the automatic control systems, a power level increase may result and, without adequate controls, could lead to fuel damage or overpressurization of the RCS. Alternatively, a power level decrease and depressurization may result. Reactor protection and safety systems are actuated to mitigate these events. The NRC staff's review covered (1) the sequence of events, (2) the analytical model used for analyses, (3) the values of parameters used in the analytical model, and (4) the results of the transient analyses. The NRC's acceptance criteria are based on (1) draft GDC-6, insofar as it requires that the reactor core be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits; and (2) draft GDC-27 and 28, insofar as they require that at least two reactivity control systems be provided and be capable of making and holding the core subcritical from any hot standby or hot operating condition sufficiently fast to prevent exceeding acceptable fuel damage limits. Specific review criteria are contained in SRP Section 15.5.1-2 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

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

Conclusion

The NRC staff has reviewed the licensee's analyses of the inadvertent operation of ECCS or malfunction that increases reactor coolant inventory and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the AFDLs 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 draft GDC-6, 27, and 28 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the inadvertent operation of ECCS or malfunction that increases reactor coolant inventory.

2.8.5.6 Decrease in Reactor Coolant Inventory

2.8.5.6.1 Inadvertent Opening of a Pressure Relief Valve

Regulatory Evaluation

The inadvertent opening of a pressure relief valve results in a reactor coolant inventory decrease and a decrease in RCS pressure. The pressure relief valve discharges into the suppression pool. Normally there is no reactor trip. The pressure regulator senses the RCS pressure decrease and partially closes the turbine control valves (TCVs) to stabilize the reactor at a lower pressure. The reactor power settles out at nearly the initial power level. The coolant inventory is maintained by the feedwater control system using water from the condensate storage tank via the condenser hotwell. The NRC staff's review covered (1) the sequence of events, (2) the analytical model used for analyses, (3) the values of parameters used in the analytical model, and (4) the results of the transient analyses. The NRC's acceptance criteria are based on (1) draft GDC-6, insofar as it requires that the reactor core be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits; and (2) draft GDC-27 and 28, insofar as they require that at least two reactivity control systems be provided and be capable of making and holding the core subcritical from any hot standby or hot operating condition sufficiently fast to prevent exceeding acceptable fuel damage limits. Specific review criteria are contained in SRP Section 15.6.1 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

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

Conclusion

The NRC staff has reviewed the licensee's analyses of the inadvertent opening of a pressure relief valve event and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the AFDLs 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 draft GDC-6, 27, and 28 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the inadvertent opening of a pressure relief valve event.

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

Regulatory Evaluation

LOCAs are postulated accidents that would result in the loss of reactor coolant from piping breaks in the RCPB at a rate in excess of the capability of the normal reactor coolant makeup system to replenish it. Loss of significant quantities of reactor coolant would prevent heat removal from the reactor core, unless the water is replenished. The reactor protection and ECCS systems are provided to mitigate these accidents. The NRC staff's review covered (1) the licensee's determination of break locations and break sizes; (2) postulated initial conditions; (3) the sequence of events; (4) the analytical model used for analyses, and calculations of the reactor power, pressure, flow, and temperature transients; (5) calculations of peak cladding temperature, total oxidation of the cladding, total hydrogen generation, changes in core geometry, and long-term cooling; (6) functional and operational characteristics of the reactor protection and ECCS systems; and (7) operator actions. The NRC's acceptance criteria are based on (1) 10 CFR § 50.46, insofar as it establishes standards for the calculation of ECCS performance and acceptance criteria for that calculated performance; (2) 10 CFR Part 50, Appendix K, insofar as it establishes required and acceptable features of evaluation models for heat removal by the ECCS after the blowdown phase of a LOCA; (3) draft GDC-40 and 42, insofar as they require that protection be provided for ESFs against the dynamic effects that might result from plant equipment failures, as well as the effects of a LOCA; and (4) draft GDC-37, 41, and 44, insofar as they require that a system to provide abundant emergency core cooling be provided so that fuel and clad damage that would interfere with the emergency core cooling function will be prevented. Specific review criteria are contained in SRP Sections 6.3 and 15.6.5 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

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

Conclusion

The NRC staff has reviewed the licensee's analyses of the LOCA events and the ECCS. The NRC staff concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and that the analyses were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection system and the ECCS will continue to ensure that the peak cladding temperature, total oxidation of the cladding, total hydrogen generation, and changes in core geometry, and long-term cooling will remain within acceptable limits. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of draft GDC-37, 40, 41, 42, and 44, and 10 CFR 50.46 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the LOCA.

2.8.5.7 Anticipated Transients Without Scrams

Regulatory Evaluation

ATWS is defined as an AOO followed by the failure of the reactor portion of the protection system specified in draft GDC-14 and 15. The regulation at 10 CFR 50.62 requires that:

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

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

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

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

Technical Evaluation

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

Conclusion

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

2.8.6 Fuel Storage

2.8.6.1 New Fuel Storage

Regulatory Evaluation

Nuclear reactor plants include facilities for the storage of new fuel. The quantity of new fuel to be stored varies from plant to plant, depending upon the specific design of the plant and the individual refueling needs. The NRC staff's review covered the ability of the storage facilities to maintain the new fuel in a subcritical array during all credible storage conditions. The review focused on the effect of changes in fuel design on the analyses for the new fuel storage facilities. The NRC's acceptance criteria are based on draft GDC-66, insofar as it requires the prevention of criticality in fuel storage systems by physical systems or processes, preferably utilizing geometrically safe configurations. Specific review criteria are contained in SRP Section 9.1.1.

Technical Evaluation

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

Conclusion

The NRC staff has reviewed the licensee's analyses related to the effect of the new fuel on the analyses for the new fuel storage facilities and concludes that the new fuel storage facilities will continue to meet the requirements of draft GDC-66 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the new fuel storage.

2.8.6.2 Spent Fuel Storage

Regulatory Evaluation

Nuclear reactor plants include storage facilities for the wet storage of spent fuel assemblies. The safety function of the spent fuel pool and storage racks is to maintain the spent fuel assemblies in a safe and subcritical array during all credible storage conditions and to provide a safe means of loading the assemblies into shipping casks. The NRC staff's review covered the effect of the proposed EPU on the criticality analysis (e.g., reactivity of the spent fuel storage array and boraflex degradation or neutron poison efficacy). The NRC's acceptance criteria are based on (1) draft GDC-40 and 42, insofar as they require that protection be provided for ESFs against the dynamic effects and missiles that might result from plant equipment failures, as well as the effects of a loss of coolant accident; and (2) draft GDC-66, insofar as it requires that criticality in the fuel storage systems be prevented by physical systems or processes, preferably by use of geometrically safe configurations. Specific review criteria are contained in SRP Section 9.1.2.

Technical Evaluation

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

Conclusion

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed EPU on the spent fuel storage capability and concludes that the licensee has adequately accounted for the effects of the proposed EPU on the spent fuel rack temperature and criticality analyses. The NRC staff also concludes that the spent fuel pool design will continue to ensure an acceptably low temperature and an acceptable degree of subcriticality following implementation of the proposed EPU. Based on this, the NRC staff concludes that the spent fuel storage facilities will continue to meet the requirements of draft GDC-40, 42, and 66 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to spent fuel storage.

[2.8.7 Additional Review Areas (Reactor Systems)]

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

INSERT 9

FOR

SECTION 3.2 - BWR TEMPLATE SAFETY EVALUATION

2.9 Source Terms and Radiological Consequences Analyses

2.9.1 Source Terms for Radwaste Systems Analyses

Regulatory Evaluation

The NRC staff reviewed the radioactive source term associated with EPUs to ensure the adequacy of the sources of radioactivity used by the licensee as input to calculations to verify that the radioactive waste management systems have adequate capacity for the treatment of radioactive liquid and gaseous wastes. The NRC staff's review included the parameters used to determine (1) the concentration of each radionuclide in the reactor coolant, (2) the fraction of fission product activity released to the reactor coolant, (3) concentrations of all radionuclides other than fission products in the reactor coolant, (4) leakage rates and associated fluid activity of all potentially radioactive water and steam systems, and (5) potential sources of radioactive materials in effluents that are not considered in the plant's **[Updated Safety Analysis Report *or* Updated Final Safety Analysis Report]** related to liquid waste management systems and gaseous waste management systems. The NRC's acceptance criteria for source terms are based on (1) 10 CFR Part 20, insofar as it establishes requirements for radioactivity in liquid and gaseous effluents released to unrestricted areas; (2) 10 CFR Part 50, Appendix I, insofar as it establishes numerical guides for design objectives and limiting conditions for operation to meet the "as low as is reasonably achievable" criterion; and (3) draft GDC-70, insofar as it requires that the plant design include means to control the release of radioactive effluents. Specific review criteria are contained in SRP Section 11.1.

Technical Evaluation

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

Conclusion

The NRC staff has reviewed the radioactive source term associated with the proposed EPU and concludes that the proposed parameters and resultant composition and quantity of radionuclides are appropriate for the evaluation of the radioactive waste management systems. The NRC staff further concludes that the proposed radioactive source term meets the requirements of 10 CFR Part 20, 10 CFR Part 50, Appendix I, and draft GDC-70. Therefore, the NRC staff finds the proposed EPU acceptable with respect to source terms.

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

2.9.2 Radiological Consequences Analyses Using Alternative Source Terms

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

Regulatory Evaluation

The NRC staff reviewed the DBA radiological consequences analyses. The radiological consequences analyses reviewed are the LOCA, fuel handling accident (FHA), control rod drop accident (CRDA), and main steamline break (MSLB). The NRC staff's review for each accident analysis included (1) the sequence of events; and (2) models, assumptions, and values of parameter inputs used by the licensee for the calculation of the total effective dose equivalent (TEDE). The NRC's acceptance criteria for radiological consequences analyses using an alternative source term are based on (1) 10 CFR 50.67, insofar as it sets standards for radiological consequences of a postulated accident, and (2) draft GDC-11, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem TEDE, as defined in 10 CFR 50.2, for the duration of the accident. Specific review criteria are contained in SRP Section 15.0.1.

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

The NRC staff reviewed the implementation of alternative source terms. The NRC's acceptance criteria for implementation of alternative source terms are based on (1) 10 CFR 50.67, insofar as it sets standards for the implementation of an alternative source term in current operating nuclear power plants; (2) 10 CFR 50.49, insofar as it requires qualification of safety-related equipment, as defined in that section, including and based on integrated radiation dose during normal and accident conditions; (3) draft GDC-11, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem TEDE, as defined in 10 CFR 50.2, for the duration of the accident; (4) Paragraph IV.E.8 of 10 CFR Part 50, Appendix E, insofar as it requires a licensee onsite technical support center and a licensee near-site emergency operations facility from which effective direction can be given and effective control can be exercised during an emergency; and (5) plant-specific licensing commitments made in response to NUREG-0737 (Items II.B.2, II.B.3, II.F.1, III.D.1.1, III.A.1.2, and III.D.3.4). Specific review criteria are contained in SRP Sections 15.0.1.

Technical Evaluation

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

Conclusion

The NRC staff has evaluated the licensee's revised accident analyses performed in support of the proposed EPU and concludes that the licensee has adequately accounted for the effects of the proposed EPU. The NRC staff further concludes that the plant site and the dose-mitigating ESFs remain acceptable with respect to the radiological consequences of postulated DBAs since, as set forth above, the calculated total effective dose equivalent (TEDE) at the exclusion area boundary (EAB), at the low population zone (LPZ) outer boundary, and in the control room meet the exposure guideline values specified in 10 CFR 50.67 and draft GDC-11, as well as applicable acceptance criteria denoted in SRP Section 15.0.1. Therefore, the NRC staff finds the licensee's proposed EPU acceptable with respect to the radiological consequences of DBAs.

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

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

[2.9.3 Additional Review Areas (Radiological Consequences Analyses)]

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

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

2.9.2 <u>Radiological Consequences of Control Rod Drop Accident</u>

[This section is not applicable because BFN has implemented an alternative source term.]

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

[This section is not applicable because BFN has implemented an alternative source term.]

2.9.4 Radiological Consequences of Main Steamline Failure Outside Containment

[This section is not applicable because BFN has implemented an alternative source term.]

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

[This section is not applicable because BFN has implemented an alternative source term.]

2.9.6 Radiological Consequences of Fuel Handling Accidents

[This section is not applicable because BFN has implemented an alternative source term.]

2.9.7 Radiological Consequences of Spent Fuel Cask Drop Accidents

[This section is not applicable because BFN has implemented an alternative source term.]

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

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

INSERT 10

FOR

SECTION 3.2 - BWR TEMPLATE SAFETY EVALUATION

2.10 Health Physics

2.10.1 Occupational and Public Radiation Doses

Regulatory Evaluation

The NRC staff conducted its review in this area to ascertain what overall effects the proposed EPU will have on both occupational and public radiation doses and to determine that the licensee has taken the necessary steps to ensure that any dose increases will be maintained as low as is reasonably achievable. The NRC staff's review included an evaluation of any increases in radiation sources and how this may affect plant area dose rates, plant radiation zones, and plant area accessibility. The NRC staff evaluated how personnel doses needed to access plant vital areas following an accident are affected. The NRC staff considered the effects of the proposed EPU on nitrogen-16 levels in the plant and any effects this increase may have on radiation doses outside the plant and at the site boundary from skyshine. The NRC staff also considered the effects of the proposed EPU on plant effluent levels and any effect this increase may have on radiation doses at the site boundary. The NRC's acceptance criteria for occupational and public radiation doses are based on 10 CFR Part 20 10 CFR 50.67, and draft GDC-11. Specific review criteria are contained in SRP Sections 12.2, 12.3,12.4, and 12.5, and other guidance provided in Matrix 10 of RS-001.

Technical Evaluation

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

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on radiation source terms and plant radiation levels. The NRC staff concludes that the licensee has taken the necessary steps to ensure that any increases in radiation doses will be maintained as low as reasonably achievable. The NRC staff further concludes that the proposed EPU meets the requirements of 10 CFR Part 20 and draft GDC-11. Therefore, the NRC staff finds the licensee's proposed EPU acceptable with respect to radiation protection and ensuring that occupational radiation exposures will be maintained as low as reasonably achievable.

[2.10.2 Additional Review Areas (Health Physics)]

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

INSERT 11

FOR

SECTION 3.2 - BWR TEMPLATE SAFETY EVALUATION

2.11 Human Performance

2.11.1 Human Factors

Regulatory Evaluation

The area of human factors deals with programs, procedures, training, and plant design features related to operator performance during normal and accident conditions. The NRC staff's human factors evaluation was conducted to ensure that operator performance is not adversely affected as a result of system changes made to implemented the proposed EPU. The NRC staff's review covered changes to operator actions, human-system interfaces, and procedures and training needed for the proposed EPU. The NRC's acceptance criteria for human factors are based on draft GDC-11, 10 CFR 50.120, 10 CFR Part 55, and the guidance in GL 82-33. Specific review criteria are contained in SRP Sections 13.2.1, 13.2.2, 13.5.2.1, and 18.0.

Technical Evaluation

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

1. Changes in Emergency and Abnormal Operating Procedures

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

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

2. Changes to Operator Actions Sensitive to Power Uprate

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

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

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

3. Changes to Control Room Controls, Displays and Alarms

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

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

4. Changes on the Safety Parameter Display System

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

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

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

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

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

Conclusion

The NRC staff has reviewed the changes to operator actions, human-system interfaces, procedures, and training required for the proposed EPU and concludes that the licensee has (1) appropriately accounted for the effects of the proposed EPU on the available time for operator actions and (2) taken appropriate actions to ensure that operator performance is not adversely affected by the proposed EPU. The NRC staff further concludes that the licensee will continue to meet the requirements of draft GDC-11, 10 CFR 50.120, and 10 CFR Part 55 following implementation of the proposed EPU. Therefore, the NRC staff finds the licensee's proposed EPU acceptable with respect to the human factors aspects of the required system changes.

[2.11.2 Additional Review Areas (Human Performance)]

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

INSERT 12

FOR

SECTION 3.2 - BWR TEMPLATE SAFETY EVALUATION

2.12 Power Ascension and Testing Plan

2.12.1 Approach to EPU Power Level and Test Plan

Regulatory Evaluation

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

Technical Evaluation

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

Conclusion

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

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

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

INSERT 13

FOR

SECTION 3.2 - BWR TEMPLATE SAFETY EVALUATION

2.13 Risk Evaluation

2.13.1 Risk Evaluation of EPU

Regulatory Evaluation

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

Technical Evaluation

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

Conclusion

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

[2.13.2 Additional Review Areas (Risk Evaluation)]

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

ENCLOSURE 3

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT UNITS 2 AND 3
DOCKET NOS. 50-260, 50-296

EXTENDED POWER UPRATE RS-001 REVISED AREAS OF REVIEW MATRIX

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		Cross-Reference to	
							BWR	PWR	PUSAR/ FUSAR	ELTR
Reactor Vessel Material Surveillance Program	All EPUs	EMCB	SRXB	5.3.1 Draft Rev. 2 April 1996	GDC-14 GDC-31 10 CFR Part 50, App. H 10 CFR 50.60	RG 1.190	2.1.1	2.1.1	3.3.1	ELTR1 5.5.1.5
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 Part 50, App. G 10 CFR 50.60	RG 1.161 RG 1.190 RG 1.99	2.1.2	2.1.2	3.3.1	ELTR1 5.5.1.5
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	N/A for BWR's	
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	10.7	ELTR1 5.11.6

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		Cross-Reference to	
							BWR	PWR	PUSAR/ FUSAR	ELTR
Reactor Coolant Pressure Boundary Materials	All EPUs	EMCB	EMEB SRXB	5.2.3 Draft Rev. 3 April 1996	GDC-1 10 CFR 50.55a GDC-4 GDC-14 GDC-31 10 CFR Part 50, App. G	RG 1.190 GL 97-01 IN 00-17s1 BL 01-01 BL 02-01 BL 02-02 Note 2* Note 3*	2.1.4	2.1.5	10.7	ELTR1 5.11.6; ELTR2 3.6.1
				4.5.1 Draft Rev. 3 April 1996	GDC-1 10 CFR 50.55a GDC-14					
				5.2.4 Draft Rev. 2 April 1996	10 CFR 50.55a					
				5.3.1 Draft Rev. 2 April 1996 5.3.3 Draft Rev. 2 April 1996 6.1.1 Draft Rev. 2 April 1996	GDC-1 10 CFR 50.55a GDC-4 GDC-14 GDC-31 10 CFR Part 50, App. G					
Leak-Before-Break	PWR EPUs	EMCB		3.6.3 Draft Aug. 1987	GDC-4	NUREG 1061 Vol. 3 Nov. 1984		2.1.6	N/A for BWR's	

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		Cross-Reference to	
							BWR	PWR	PUSAR/ FUSAR	ELTR
Protective Coating Systems (Paints) - Organic Materials	All EPUs	EMCB		6.1.2 Draft Rev. 3 April 1996	10 CFR Part 50, App. B RG 1.54		2.1.5	2.1.7	4.2.5	N/A
Effect of EPU on Flow-Accelerated Corrosion	All EPUs	EMCB				Note 4*	2.1.6	2.1.8	3.11.3	ELTR1 5.10.10; ELTR2 3.6.1
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	N/A for BWR's	
Steam Generator Blowdown System	PWR EPUs	EMCB		10.4.8 Draft Rev. 3 April 1996	GDC-14			2.1.10	N/A for BWR's	
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	N/A for BWR's	
Reactor Water Cleanup System	BWR EPUs	EMCB		5.4.8 Draft Rev. 3 April 1996	GDC-14 GDC-60 GDC-61		2.1.7		3.5, 3.10, 3.11, 10.1	ELTR1 5.6.6, 5.11.8, J.2.3.4

Notes:

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

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

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

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

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

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		Cross-Reference to	
							BWR	PWR	PUSAR/ FUSAR	ELTR
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	10.1	ELTR1 5.11.8, Appendix K

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		Cross-Reference to	
							BWR	PWR	PUSAR/ FUSAR	ELTR
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	2.5, 3.3, 3.4, 3.5, 3.6, 3.7, 3.11	ELTR1 5.5.1, 5.5.2, 5.6.2, 5.6.3, 5.10.10, Appendix I, J.2.3, Appendix K
				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		Cross-Reference to	
							BWR	PWR	PUSAR/ FUSAR	ELTR
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	3.3.3, 3.3.4, 3.3.5 BFN Note	ELTR1 5.5.1, 5.5.1.1, 5.5.1.2, 5.5.1.3
				3.9.2 Draft Rev. 3 April 1996	GDC-1 GDC-2 GDC-4	IN 95-016 IN 02-026				
				3.9.3 Draft Rev. 2 April 1996	10 CFR 50.55a GDC-1 GDC-2 GDC-4	IN 96-049 GL 96-06				
				3.9.5 Draft Rev. 3 April 1996	10 CFR 50.55a GDC-1 GDC-2 GDC-4 GDC-10	IN 02-026 Note 1*				

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Cross-Reference to	
							BWR	PWR	PUSAR/ FUSAR	ELTR
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	3.1, 3.7, 4.1.3, 4.1.4, 4.1.6, 4.2	ELTR1 5.6.4, 5.6.7, 5.6.8, 5.6.9, J.2.3
				3.9.6 Draft Rev. 3 April 1996	GDC-1 GDC-37 GDC-40 GDC-43 GDC-46 GDC-54 10 CFR 50.55a(f)	GL 89-10 GL 95-07 GL 96-05 IN 97-090 IN 96-048s1 IN 96-048 IN 96-003 RIS 00-003 RIS 01-015 RG 1.147 RG 1.175 DG 1089 DG 1091				
Seismic and Dynamic Qualification of Mechanical and Electrical Equipment	All EPUs	EMEB	EEIB	3.10 Draft Rev. 3 April 1996	GDC-1 GDC-2 GDC-4 GDC-14 GDC-30 10 CFR Part 100, App. A 10 CFR Part 50, App. B USI A-46		2.2.5	2.2.5	10.1	ELTR1 5.11.8, Appendix K

Notes:

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

reactor internal components such as steam dryers and separators, and the jet pump sensing lines that are affected by the increased steam and feedwater flow for EPU conditions. In addition, the staff is evaluating the need to address potential adverse effects on other plant components from the increased steam and feedwater flow under EPU conditions.

BROWNS FERRY NOTES - MATRIX 2

SE 2.2.3 BFN NOTE, Reactor Pressure Vessel Internals and Core Supports: Additional information is provided by Enclosures 9 & 10 to the initial License Amendment Request.

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		Cross-Reference to	
							BWR	PWR	PUSAR/ FUSAR	ELTR
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	10.3.1 (FUSAR 10.3)	ELTR1 5.11.2
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	6.1.1 BFN Note	ELTR1 5.10.6
				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	6.1.2	ELTR1 5.10.6
				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		Cross-Reference to	
							BWR	PWR	PUSAR/ FUSAR	ELTR
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	6.2	ELTR1 5.10.6
				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	9.3.2	ELTR1 5.11.7
				8.2, App. B Draft Rev. 4 April 1996	10 CFR 50.63					

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

BROWNS FERRY NOTES - MATRIX 3

SE 2.3.2 BFN NOTE, Offsite Power System: Additional information is provided by Enclosure 11 to the initial License Amendment Request.

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		Cross-Reference to	
							BWR	PWR	PUSAR/ FUSAR	ELTR
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	5	ELTR1 5.8, Appendix F
Engineered Safety Features Systems	All EPUs	EEIB		7.3 Rev. 4 June 1997			2.4.1	2.4.1	5	ELTR1 5.8, Appendix F
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	5	ELTR1 5.8, Appendix F
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		2.4.1	2.4.1	5	ELTR1 5.8, Appendix F

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		Cross-Reference to	
							BWR	PWR	PUSAR/ FUSAR	ELTR
Diverse I&C Systems	All EPUs	EEIB		7.8 Rev. 4 June 1997	GDC-19 GDC-24		2.4.1	2.4.1	5	ELTR1 5.8, Appendix F
General guidance for use of other SRP Sections related to I&C	All EPUs	EEIB		7.0 Rev. 4 June 1997						

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		Cross-Reference to	
							BWR	PWR	PUSAR/ FUSAR	ELTR
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	10.1, 10.2	ELTR1 5.11.8
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	BFN Note	NA
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	BFN Note	NA
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	BFN Note	NA

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		Cross-Reference to	
							BWR	PWR	PUSAR/ FUSAR	ELTR
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	BFN Note	NA
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	7.1	ELTR1 5.10.1
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	10.1, 10.2	ELTR1 5.11.8
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 Part 50, App. R GDC-3 GDC-5	Note 1*	2.5.1.4	2.5.1.4	6.7	ELTR1 5.11.1
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	N/A for BWR's	
Fission Product Control Systems and Structures	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB	EMCB	6.5.3 Rev. 2 July 1981	GDC-41		2.5.2.1	2.5.3.1	4.4, 4.5, 9.2	ELTR1 5.4

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		Cross-Reference to	
							BWR	PWR	PUSAR/ FUSAR	ELTR
Main Condenser Evacuation System	EPUs for which the main condenser evacuation system is modified	SPLB		10.4.2 Rev. 2 July 1981	GDC-60 GDC-64		2.5.2.2	2.5.3.2	7.2	ELTR1 J.2.3
Turbine Gland Sealing System	EPUs for which the turbine gland sealing system is modified	SPLB		10.4.3 Rev. 2 July 1981	GDC-60 GDC-64		2.5.2.3	2.5.3.3	NA	NA
Main Steam Isolation Valve Leakage Control System	BWR EPU that affect the amount of valve leakage that is assumed and resultant dose consequences.	SPLB		6.7 Rev. 2 July 1981	GDC-54		2.5.2.4		4.6	NA
Spent Fuel Pool Cooling and Cleanup System	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB	EMCB	9.1.3 Rev. 1 July 1981	GDC-5 GDC-44 GDC-61	Note 2*	2.5.3.1	2.5.4.1	6.3.1	ELTR1 5.10.8; ELTR2 S1 V1 4.1.7
Station Service Water System	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB		9.2.1 Rev. 4 June 1985	GDC-4 GDC-5 GDC-44	GL 89-13 and Suppl. 1 GL 96-06 and Suppl. 1	2.5.3.2	2.5.4.2	6.4.1 (FUSAR 6.4)	ELTR1 5.10.4
Reactor Auxiliary Cooling Water Systems	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB		9.2.2 Rev. 3 June 1986	GDC-4 GDC-5 GDC-44	GL 89-13 and Suppl. 1 GL 96-06 and Suppl. 1	2.5.3.3	2.5.4.3	BFN Note	NA

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		Cross-Reference to	
							BWR	PWR	PUSAR/ FUSAR	ELTR
Ultimate Heat Sink	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB		9.2.5 Rev. 2 July 1981	GDC-5 GDC-44		2.5.3.4	2.5.4.4	6.4.5 (FUSAR 6.4)	ELTR1 5.10.4
Auxiliary Feedwater System	PWR EPUs except where the application demonstrates that previous analysis is bounding	SPLB		10.4.9 Rev. 2 July 1981	GDC-4 GDC-5 GDC-19 GDC-34 GDC-44			2.5.4.5	N/A for BWR's	
Main Steam Supply System	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB		10.3 Rev. 3 April 1984	GDC-4 GDC-5 GDC-34		2.5.4.1	2.5.5.1	3.5.2, 3.6, 3.7, 3.11.4	ELTR1 5.5.2, 5.10.10, J.2.3
Main Condenser	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB		10.4.1 Rev. 2 July 1981	GDC-60		2.5.4.2	2.5.5.2	7.2	ELTR 5.10.4
Turbine Bypass System	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB		10.4.4 Rev. 2 July 1981	GDC-4 GDC-34		2.5.4.3	2.5.5.3	7.3	ELTR1 F.2.2, J.2.3
Condensate and Feedwater System	All EPUs except where the application demonstrates that previous analysis is bounding	SPLB		10.4.7 Rev. 3 April 1984	GDC-4 GDC-5 GDC-44		2.5.4.4	2.5.5.4	7.4	ELTR1 5.10.3

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		Cross-Reference to	
							BWR	PWR	PUSAR/ FUSAR	ELTR
Gaseous Waste Management Systems	EPUs that impact the level of fission products in the reactor coolant system, or the amount of gaseous waste	SPLB	IEPB	11.3 Draft Rev. 3 April 1996	10 CFR 20.1302 GDC-3 GDC-60 GDC-61 10 CFR Part 50, App. I		2.5.5.1	2.5.6.1	8.2, 8.6	ELTR1 5.10.9, J.2.2
Liquid Waste Management Systems	EPUs that impact the level of fission products in the reactor coolant system, or the amount of liquid waste	SPLB	IEPB	11.2 Draft Rev. 3 April 1996	10 CFR 20.1302 GDC-60 GDC-61 10 CFR Part 50, App. I		2.5.5.2	2.5.6.2	8.1, 8.6	ELTR1 5.10.9
Solid Waste Management Systems	EPUs that impact the level of fission products in the reactor coolant system, or the amount of solid waste	SPLB	IEPB	11.4 Draft Rev. 3 April 1996	10 CFR 20.1302 GDC-60 GDC-63 GDC-64 10 CFR Part 71		2.5.5.3	2.5.6.3	8.1	ELTR1 5.10.9
Emergency Diesel Engine Fuel Oil Storage and Transfer System	EPUs that result in higher EDG electrical demands	SPLB		9.5.4 Rev. 2 July 1981	GDC-4 GDC-5 GDC-17		2.5.6.1	2.5.7.1	BFN Note	NA
Light Load Handling System (Related to Refueling)	EPUs except where the application demonstrates that previous analysis is bounding	SPLB	SPSB	9.1.4 Rev. 2 July 1981	GDC-61 GDC-62		2.5.6.2	2.5.7.2	BFN Note	NA

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

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

BROWNS FERRY NOTES - MATRIX 5

SE 2.5.1.1.2 BFN NOTE, Equipment and Floor Drainage System: This system is not dependent upon power level.

SE 2.5.1.1.3 BFN NOTE, Circulating Water System: Circulating water system flowrate or capacity is not increased for BFN EPU.

SE 2.5.1.2.1 BFN NOTE, Internally Generated Missiles (Outside Containment): The BFN internally generated missiles evaluations are not impacted by BFN EPU.

SE 2.5.1.2.1 BFN NOTE, Internally Generated Missiles (Inside Containment): The BFN internally generated missiles evaluations are not impacted by BFN EPU.

SE 2.5.3.3 BFN NOTE, Reactor Auxiliary Cooling Water Systems: BFN does not have a Reactor Auxiliary Cooling Water System

SE 2.5.6.1 BFN NOTE, Emergency Diesel Engine Fuel Oil Storage and Transfer System: There is no increase in emergency power loads for BFN EPU.

SE 2.5.6.2 BFN NOTE, Light Load Handling System (Related to Refueling): The fuel handling and storage system is not impacted by BFN EPU.

MATRIX 6

SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

Containment Review Considerations

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Cross-Reference to	
							BWR	PWR	PUSAR/ FUSAR	ELTR
PWR Dry Containments, Including Subatmospheric Containments	EPUs for PWR plants with dry containments (including subatmospheric containments) except where the application demonstrates that previous analysis is bounding	SPSB		6.2.1 Rev. 2 July 1981 6.2.1.1.A Rev. 2 July 1981	GDC-13 GDC-16 GDC-38 GDC-50 GDC-64			2.6.1	N/A for BWR's	
Ice Condenser Containments	EPUs for PWR plants with ice condenser containments except where the application demonstrates that previous analysis is bounding	SPSB		6.2.1 Rev. 2 July 1981 6.2.1.1.B Rev. 2 July 1981	GDC-13 GDC-16 GDC-38 GDC-50 GDC-64			2.6.1	N/A for BWR's	
Pressure-Suppression Type BWR Containments	EPUs for BWR plants with pressure-suppression containments except where the application demonstrates that previous analysis is bounding	SPSB		6.2.1 Rev. 2 July 1981 6.2.1.1.C Rev. 6 Aug. 1984	GDC-4 GDC-13 GDC-16 GDC-50 GDC-64		2.6.1		4.1	ELTR1 5.10.2, Appendix G

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		Cross-Reference to	
							BWR	PWR	PUSAR/ FUSAR	ELTR
Subcompartment Analysis	All EPUs except where the application demonstrates that previous analysis is bounding	SPSB		6.2.1 Rev. 2 July 1981 6.2.1.2 Rev. 2 July 1981	GDC-4 GDC-50		2.6.2	2.6.2	4.1.2.3	ELTR1 5.10.2, G.2.3
Mass and Energy Release Analysis for Postulated Loss-of-Coolant	All EPUs except where the application demonstrates that previous analysis is bounding	SPSB		6.2.1 Rev. 2 July 1981 6.2.1.3 Rev. 1 July 1981	GDC-50 10 CFR Part 50, App. K		2.6.3.1	2.6.3.1	4.1	ELTR1 5.10.2, Appendix G
Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures	PWR EPUs except where the application demonstrates that previous analysis is bounding	SPSB		6.2.1 Rev. 2 July 1981 6.2.1.4 Rev. 1 July 1981	GDC-50			2.6.3.2	N/A for BWR's	
Combustible Gas Control In Containment	EPUs that impact hydrogen release assumptions	SPSB		6.2.5 Rev. 2 July 1981	10 CFR 50.44 10 CFR 50.46 GDC-5 GDC-41 GDC-42 GDC-43		2.6.4	2.6.4	4.7	ELTR2 S1 V1 3.5

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		Cross-Reference to	
							BWR	PWR	PUSAR/ FUSAR	ELTR
Containment Heat Removal	All EPUs except where the application demonstrates that previous analysis is bounding	SPSB		6.2.2 Rev. 4 Oct. 1985	GDC-38	DG-1107	2.6.5	2.6.5	4.1, 4.2.5	ELTR1 5.10.2, Appendix G; ELTR2 S1 V1 4.1.8.5
Secondary Containment Functional Design	EPUs that affect the pressure and temperature response, or draw-down time of the secondary containment	SPSB		6.2.3 Rev. 2 July 1981	GDC-4 GDC-16		2.6.6		10.1, 10.2, 10.3, 9.2	ELTR1 5.11.2, 5.11.8
Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies	PWR EPUs except where the application demonstrates that previous analysis is bounding	SPSB	SRXB	6.2.1 Rev. 2 July 1981; 6.2.1.5 Rev. 2 July 1981	10 CFR 50.46 10 CFR Part 50, App. K			2.6.6	N/A for BWR's	

MATRIX 7

SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

Habitability, Filtration, and Ventilation

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Cross-Reference to	
							BWR	PWR	PUSAR/ FUSAR	ELTR
Control Room Habitability System	All EPUs except where the application demonstrates that previous analysis is bounding	SPSB		6.4 Draft Rev. 3 April 1996	GDC-4 GDC-19	Note 1* Note 2*	2.7.1	2.7.1	4.4	ELTR1 5.4
ESF Atmosphere Cleanup System	All EPUs except where the application demonstrates that previous analysis is bounding	SPSB		6.5.1 Rev. 2 July 1981	GDC-19 GDC-41 GDC-61 GDC-64		2.7.2	2.7.2	4.4, 4.5	ELTR1 5.4
Control Room Area Ventilation System	All EPUs except where the application demonstrates that previous analysis is bounding	SPSB		9.4.1 Rev. 2 July 1981	GDC-4 GDC-19 GDC-60		2.7.3	2.7.3	BFN Note	NA
Spent Fuel Pool Area Ventilation System	All EPUs except where the application demonstrates that previous analysis is bounding	SPSB		9.4.2 Rev. 2 July 1981	GDC-60 GDC-61		2.7.4	2.7.4	BFN Note	NA

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		Cross-Reference to	
							BWR	PWR	PUSAR/ FUSAR	ELTR
Auxiliary and Radwaste Area Ventilation System	All EPUs except where the application demonstrates that previous analysis is bounding	SPSB		9.4.3 Rev. 2 July 1981	GDC-60		2.7.5	2.7.5	BFN Note	NA
Turbine Area Ventilation System	All EPUs except where the application demonstrates that previous analysis is bounding	SPSB		9.4.4 Rev. 2 July 1981	GDC-60		2.7.5	2.7.5	6.6	ELTR1 J.2.3
ESF Ventilation System	All EPUs except where the application demonstrates that previous analysis is bounding	SPSB		9.4.5 Rev. 2 July 1981	GDC-4 GDC-17 GDC-60		2.7.6	2.7.6	BFN Note	NA

Notes:

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

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

BROWNS FERRY NOTES - MATRIX 7

SE 2.7.3 BFN NOTE, Control Room Area Ventilation System: No EPU effect.

SE 2.7.4 BFN NOTE, Spent Fuel Pool Area Ventilation System: BFN does not have a separate Spent Fuel Pool Area Ventilation System installed

SE 2.7.5 BFN NOTE, Auxiliary and Radwaste Area Ventilation System: No EPU effect.

SE 2.7.6 BFN NOTE, ESF Ventilation System: There are no changes to the ESF ventilation systems as a result of BFN EPU.

MATRIX 8

SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

Reactor Systems

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Cross-Reference to	
							BWR	PWR	PUSAR/ FUSAR	ELTR
Fuel System Design	All EPUs	SRXB		4.2 Draft Rev. 3 April 1996	10 CFR 50.46 GDC-10 GDC-27 GDC-35	Note 1* Note 2*	2.8.1	2.8.1	2	ELTR1 5.7
Nuclear Design	All EPUs	SRXB		4.3 Draft Rev. 3 April 1996	GDC-10 GDC-11 GDC-12 GDC-13 GDC-20 GDC-25 GDC-26 GDC-27 GDC-28	RG 1.190 GSI 170 IN 97-085	2.8.2	2.8.2	2, 3.3.1	ELTR1 5.5.1.5, 5.7
Thermal and Hydraulic Design	All EPUs	SRXB		4.4 Draft Rev. 2 April 1996	GDC-10 GDC-12	Note 3*	2.8.3	2.8.3	2, 3.3.3	ELTR1 5.3.3, 5.7

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Cross-Reference to	
							BWR	PWR	PUSAR/ FUSAR	ELTR
Functional Design of Control Rod Drive System	All EPUs	SRXB	SPLB	4.6 Draft Rev. 2 April 1996	GDC-4 GDC-23 GDC-25 GDC-26 GDC-27 GDC-28 GDC-29 10 CFR 50.62(c)(3)		2.8.4.1	2.8.4.1	2.5	ELTR1 5.6.3, J.2.3.3
Overpressure Protection during Power Operation	All EPUs	SRXB		5.2.2 Draft Rev. 3 April 1996	GDC-15 GDC-31	Note 4*	2.8.4.2	2.8.4.2	3.2	ELTR1 5.5.1.4, Appendix E; ELTR2 3.8
Overpressure Protection during Low Temperature Operation	PWR EPUs	SRXB		5.2.2 Draft Rev. 3 April 1996	GDC-15 GDC-31			2.8.4.3	N/A for BWR's	
Reactor Core Isolation Cooling System	BWR EPUs	SRXB		5.4.6 Draft Rev. 4 April 1996	GDC-4 GDC-5 GDC-29 GDC-33 GDC-34 GDC-54 10 CFR 50.63		2.8.4.3		3.8, 9.1.3, 9.3.2	ELTR1 5.6.7; ELTR2 S1 V1 4.2.2
Residual Heat Removal System	All EPUs	SRXB		5.4.7 Draft Rev. 4 April 1996	GDC-4 GDC-5 GDC-19 GDC-34	Note 5*	2.8.4.4	2.8.4.4	3.9	ELTR1 5.6.4, J.2.3.1

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		Cross-Reference to	
							BWR	PWR	PUSAR/ FUSAR	ELTR
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 Part 50, App. K	Note 6*	2.8.5.6.2	2.8.5.6.3	4.2	ELTR1 5.6, J.2.3.1
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 10*	2.8.4.5		6.5	ELTR1 5.6.5
Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve	All EPUs	SRXB		15.1.1-4 Draft Rev. 2 April 1996	GDC-10 GDC-15 GDC-20 GDC-26	Note 7*	2.8.5.1	2.8.5.1.1	9.1, Table 1-3	ELTR1 5.3.2, Appendix E
Steam System Piping Failures Inside and Outside of Containment	PWR EPUs	SRXB		15.1.5 Draft Rev. 3 April 1996	GDC-27 GDC-28 GDC-31 GDC-35	Note 7*		2.8.5.1.2	N/A for BWR's	
Loss of External Load; Turbine Trip, Loss of Condenser Vacuum; Closure of Main Steam Isolation Valve (BWR); and Steam Pressure Regulator Failure (Closed)	All EPUs	SRXB		15.2.1-5 Draft Rev. 2 April 1996	GDC-10 GDC-15 GDC-26	Note 7*	2.8.5.2.1	2.8.5.2.1	9.1, Table 1-3	ELTR1 5.3.2, Appendix E

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		Cross-Reference to	
							BWR	PWR	PUSAR/ FUSAR	ELTR
Loss of Nonemergency AC Power to the Station Auxiliaries	All EPUs	SRXB		15.2.6 Draft Rev. 2 April 1996	GDC-10 GDC-15 GDC-26	Note 7*	2.8.5.2.2	2.8.5.2.2	9.1, Table 1-3	ELTR1 5.3.2, Appendix E
Loss of Normal Feedwater Flow	All EPUs	SRXB	EEIB	15.2.7 Draft Rev. 2 April 1996	GDC-10 GDC-15 GDC-26	Note 7*	2.8.5.2.3	2.8.5.2.3	9.1, Table 1-3	ELTR1 5.3.2, Appendix E
Feedwater System Pipe Breaks Inside and Outside Containment	PWR EPUs	SRXB	EEIB	15.2.8 Draft Rev. 2 April 1996	GDC-27 GDC-28 GDC-31 GDC-35	Note 7*		2.8.5.2.4	N/A for BWR's	
Loss of Forced Reactor Coolant Flow Including Trip of Pump Motor and Flow Controller Malfunctions	All EPUs	SRXB		15.3.1-2 Draft Rev. 2 April 1996	GDC-10 GDC-15 GDC-26	Note 7*	2.8.5.3.1	2.8.5.3.1	9.1, Table 1-3	ELTR1 5.3.2, Appendix E
Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break	All EPUs	SRXB		15.3.3-4 Draft Rev. 3 April 1996	GDC-27 GDC-28 GDC-31	Note 7*	2.8.5.3.2	2.8.5.3.2	9.1, Table 1-3	ELTR1 5.3.2, Appendix E
Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition	All EPUs	SRXB		15.4.1 Draft Rev. 3 April 1996	GDC-10 GDC-20 GDC-25	Note 7*	2.8.5.4.1	2.8.5.4.1	9.1, Table 1-3	ELTR1 5.3.2, Appendix E
Uncontrolled Control Rod Assembly Withdrawal at Power	All EPUs	SRXB		15.4.2 Draft Rev. 3 April 1996	GDC-10 GDC-20 GDC-25	Note 7*	2.8.5.4.2	2.8.5.4.2	9.1, Table 1-3	ELTR1 5.3.2, Appendix E

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		Cross-Reference to	
							BWR	PWR	PUSAR/ FUSAR	ELTR
Control Rod Misoperation (System Malfunction or Operator Error)	PWR EPUs	SRXB		15.4.3 Draft Rev. 3 April 1996	GDC-10 GDC-20 GDC-25	Note 7*		2.8.5.4.3	N/A for BWR's	
Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate	All EPUs	SRXB		15.4.4-5 Draft Rev. 2 April 1996	GDC-10 GDC-15 GDC-20 GDC-26 GDC-28	Note 7*	2.8.5.4.3	2.8.5.4.4	9.1, Table 1-3	ELTR1 5.3.2, Appendix E
Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant	PWR EPUs	SRXB		15.4.6 Draft Rev. 2 April 1996	GDC-10 GDC-15 GDC-26	Note 7*		2.8.5.4.5	N/A for BWR's	
Spectrum of Rod Ejection Accidents	PWR EPUs	SRXB		15.4.8 Draft Rev. 3 April 1996	GDC-28	Note 7*		2.8.5.4.6	N/A for BWR's	
Spectrum of Rod Drop Accidents	BWR EPUs	SRXB		15.4.9 Draft Rev. 3 April 1996	GDC-28	Note 7*	2.8.5.4.4		BFN Note	NA
Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory	All EPUs	SRXB		15.5.1-2 Draft Rev. 2 April 1996	GDC-10 GDC-15 GDC-26	Note 7* Note 8*	2.8.5.5	2.8.5.5	9.1, Table 1-3	ELTR1 5.3.2, Appendix E

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		Cross-Reference to	
							BWR	PWR	PUSAR/ FUSAR	ELTR
Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve or a BWR Pressure Relief Valve	All EPUs	SRXB		15.6.1 Draft Rev. 2 April 1996	GDC-10 GDC-15 GDC-26	Note 7*	2.8.5.6.1	2.8.5.6.1	9.1, Table 1-3	ELTR1 5.3.2, Appendix E
Steam Generator Tube Rupture	PWR EPUs	SRXB		15.6.3 Draft Rev. 3 April 1996	Note 7*	Note 7*		2.8.5.6.2	N/A for BWR's	
Loss-of Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary	All EPUs	SRXB		15.6.5 Draft Rev. 3 April 1996	GDC-35 10 CFR 50.46	Note 7* Note 9*	2.8.5.6.2	2.8.5.6.3	4.3	ELTR1 5.3.1, Appendix D
Anticipated Transient Without Scram	All EPUs	SRXB				Note 7* Note 10*	2.8.5.7	2.8.5.7	6.5, 9.3.1	ELTR1 5.3.4, L.3
New Fuel Storage	EPU applications that request approval for new fuel design.	SRXB		9.1.1 Draft Rev. 3 April 1996	GDC-62		2.8.6.1	2.8.6.1	BFN Note	NA
Spent Fuel Storage	EPU applications that request approval for new fuel design.	SRXB		9.1.2 Draft Rev. 4 April 1996	GDC-4 GDC-62		2.8.6.2	2.8.6.2	BFN Note	NA

Notes:

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

2. The current acceptance criteria for fuel damage for reactivity insertion accidents (RIAs) need revision per Research Information Letter No. 174, "Interim Assessment of Criteria for Analyzing Reactivity Accidents at High Burnup." The Office of Nuclear Regulatory Research is conducting confirmatory research on RIAs and the Office of Nuclear Reactor Regulation

is discussing the issue of fuel damage criteria with the nuclear power industry as part of the industry's proposal to increase future fuel burnup limits. In the interim, current methods for assessing fuel damage in RIAs are considered acceptable based on the NRC staff's understanding of actual fuel performance, as shown in three-dimensional kinetic calculations which indicate acceptably low fuel cladding enthalpy.

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

4. The review also covers the determination of allowable power levels with inoperable main steam safety valves.

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

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

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

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

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

   - For PWR plants with both a diverse scram system (DSS) and ATWS mitigation system actuation circuitry (AMSAC), the staff will not review ATWS for EPUs.
   - For PWR plants where a DSS is not specifically required by 10 CFR 50.62, a review is conducted to verify that the consequences of an ATWS are acceptable. The acceptance criteria is that the peak primary system pressure should not exceed the ASME Service Level C limit of 3200 psig. The peak ATWS pressure is primarily a function of the moderator temperature coefficient and the primary system relief capacity.

- For BWR plants, the review is conducted to ensure that the licensee has appropriately accounted for changes in analyses due to the uprated power level and confirm that required equipment, such as the standby liquid control system (SLCS) pumps, can deliver required flowrates. The review will also cover the SLCS relief valve margin. In addition, a review is conducted to ensure that SLCS flow can be injected at the assumed time without lifting bypass relief valves during the limiting ATWS.

BROWNS FERRY NOTES - MATRIX 8

SE 2.8.5.4.4 BFN NOTE, Spectrum of Rod Drop Accidents: The GE methodology for the CRDA analysis is based on BPWS as described in UFSAR Section 14.6.2 and is unchanged for BFN EPU. The Framatome methodology for the CRDA analysis was first applied to BFN for Unit 3 to support the Spring 2004 reload. This methodology is unchanged for BFN EPU.

SE 2.8.6.1 BFN NOTE, New Fuel Storage: The BFN EPU submittal does not request approval for new fuel design. NRC Amendment Nos. 284 (Unit 2) and 242 (Unit 3) address Technical Specification changes that were required for Framatome fuel storage at BFN.

SE 2.8.6.2 BFN NOTE, Spent Fuel Storage: The BFN EPU submittal does not request approval for new fuel design. NRC Amendment Nos. 284 (Unit 2) and 242 (Unit 3) address Technical Specification changes that were required for Framatome fuel storage at BFN.

MATRIX 9

SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

Source Terms and Radiological Consequences Analyses

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Cross-Reference to	
							BWR	PWR	PUSAR/ FUSAR	ELTR
Source Terms for Input into Radwaste Management Systems Analyses	All EPUs	SPSB		11.1 Draft Rev. 3 April 1996	10 CFR Part 20 10 CFR Part 50, App. I GDC-60		2.9.1	2.9.1	8.1, 8.2, 8.4	ELTR1 5.4
Radiological Consequence Analyses Using Alternative Source Terms	EPUs that utilize alternative source term	SPSB	EEIB EMCB EMEB IEPB SPLB SRXB	15.0.1 Rev. 0 July 2000	10 CFR 50.67 GDC-19 10 CFR 50.49 10 CFR Part 51 10 CFR Part 50, App. E NUREG-0737		2.9.2	2.9.2	9.2	ELTR1 5.4
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 Part 100	Notes 4, 5, 6, 7, 27*		2.9.2	N/A for BWR's	
				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		Cross-Reference to	
							BWR	PWR	PUSAR/ FUSAR	ELTR
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 Part 100	Notes 5, 8, 9, 27*		2.9.3	N/A for BWR's	
				6.4 Draft Rev. 3 April 1996	GDC-19	Notes 1, 2, 3, 28, 29*				
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 Part 100	Notes 4, 21, 22, 27*		2.9.4	N/A for BWR's	
				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 Part 100	Notes 9, 10, 27*	2.9.2		BFN Note	
				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 Part 100		2.9.3	2.9.5	BFN Note	
				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		Cross-Reference to	
							BWR	PWR	PUSAR/ FUSAR	ELTR
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 Part 100	Notes 4, 13, 14, 15, 27*		2.9.6	N/A for BWR's	
				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 Part 100	Note 27*	2.9.4		BFN Note	
				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 Part 100	Notes 4, 23, 24, 25, 26, 27*	2.9.5	2.9.7	BFN Note	
				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 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 Part 100	Notes 11, 27*	2.9.5	2.9.7	BFN Note	
				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		Cross-Reference to	
							BWR	PWR	PUSAR/ FUSAR	ELTR
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 Part 100	Notes 9, 12, 27*	2.9.5		BFN Note	
				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 Part 100 GDC-61	Notes 4, 5, 18, 19, 20, 27*	2.9.6	2.9.8	BFN Note	
				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 Part 100 GDC-61	Notes, 5, 16, 17, 8, 18, 27*	2.9.7	2.9.9	BFN Note	
				6.4 Draft Rev. 3 April 1996	GDC-19	Notes 1, 2, 3, 28, 29*				

Notes:

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

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

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

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

5. NUREG-1465 should not be used.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

The above statements should be replaced with the following:

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

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

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

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

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

BROWNS FERRY NOTES - MATRIX 9

SE 2.9.2 BFN NOTE, Radiological Consequences of Control Rod Drop Accident: BFN's radiological consequences analyses are based on an alternative source term.

SE 2.9.3 BFN NOTE, Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment: BFN's radiological consequences analyses are based on an alternative source term.

SE 2.9.4 BFN NOTE, Radiological Consequences of Main Steamline Failure Outside Containment for a BWR: BFN's radiological consequences analyses are based on an alternative source term.

SE 2.9.5 BFN NOTE, Radiological Consequences of a Design Basis Loss-Of-Coolant-Accident Including Containment Leakage Contribution: BFN's radiological consequences analyses are based on an alternative source term.

SE 2.9.5 BFN NOTE, Radiological Consequences of a Design Basis Loss-Of-Coolant-Accident: Leakage from ESF Components Outside Containment: BFN's radiological consequences analyses are based on an alternative source term.

SE 2.9.5 BFN NOTE, Radiological Consequences of a Design Basis Loss-Of-Coolant-Accident: Leakage from Main Steam Isolation Valves: BFN's radiological consequences analyses are based on an alternative source term.

SE 2.9.6 BFN NOTE, Radiological Consequences of Fuel Handling Accidents: BFN's radiological consequences analyses are based on an alternative source term.

SE 2.9.7 BFN NOTE, Radiological Consequences of Spent Fuel Cask Drop Accidents: BFN's radiological consequences analyses are based on an alternative source term.

MATRIX 10

SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

Health Physics

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Cross-Reference to	
							BWR	PWR	PUSAR/ FUSAR	ELTR
Radiation Sources	All EPUs	IEPB		12.2 Draft Rev. 3 April 1996	10 CFR Part 20		2.10.1	2.10.1	8.3, 8.4	ELTR1 5.4
Radiation Protection Design Features	All EPUs	IEPB		12.3-4 Draft Rev. 3 April 1996	10 CFR Part 20 GDC-19	Note 1*	2.10.1	2.10.1	8.3, 8.4, 8.5, 8.6	ELTR1 5.4
Operational Radiation Protection Program	All EPUs	IEPB		12.5 Draft Rev. 3 April 1996	10 CFR Part 20	Note 2* Note 3*	2.10.1	2.10.1	8.5, 8.6	ELTR1 5.4

Notes:

1. Regulatory Guide 8.12, "Criticality Accident Alarm Systems" has been withdrawn and should not be used.
2. Regulatory Guide 8.3, "Film Badge Performance Criteria" has been withdrawn and should not be used.
3. Regulatory Guide 8.14, "Personnel Neutron Dosimeters" has been withdrawn and should not be used.

MATRIX 11

SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

Human Performance

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Cross-Reference to	
							BWR	PWR	PUSAR/ FUSAR	ELTR
Reactor Operator Training	All EPUs	IROB		13.2.1* Draft Rev. 2 Dec. 2002	Specific review questions are provided in the template safety evaluations.		2.11	2.11	10.6	ELTR1 5.11.5
Training for Non-Licensed Plant Staff	All EPUs	IROB		13.2.2* Draft Rev. 2 Dec. 2002	Specific review questions are provided in the template safety evaluations.		2.11	2.11	10.6	N/A
Operating and Emergency Operating Procedures	All EPUs	IROB	SPLB SPSB SRXB	13.5.2.1* Draft Rev. 1 Dec. 2002	Specific review questions are provided in the template safety evaluations.		2.11	2.11	11.1.2	ELTR2 2.3
Human Factors Engineering	All EPUs	IROB		18.0** Draft Rev. 0 April 1996	Specific review questions are provided in the template safety evaluations.		2.11	2.11	10.6	N/A

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

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

MATRIX 12

SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

Power Ascension and Testing Plan

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Cross-Reference to	
							BWR	PWR	PUSAR/ FUSAR	ELTR
Power Ascension and Testing	All EPUs	IEPB	EEIB EMCB EMEB IROB SPLB SPSB SRXB	14.2.1* Draft Rev. 0 Dec. 2002	Entire Section		2.12	2.12	10.4 BFN Note	ELTR1 5.11.9, L.2

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

BROWNS FERRY NOTES - MATRIX 12

SE 2.12 BFN NOTE, Power Ascension and Testing: Additional information was provided by Enclosure 8 to the initial License Amendment Request.

MATRIX 13

SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

Risk Evaluation

Areas of Review	Applicable to	Primary Review Branch	Secondary Review Branch(es)	SRP Section Number	Focus of SRP Usage	Other Guidance	Template Safety Evaluation Section Number		Cross-Reference to	
							BWR	PWR	PUSAR/ FUSAR	ELTR
Risk Evaluation	All EPUs	SPSB				Note 1* RG 1.174 RIS 2001-02	2.13	2.13	10.5	ELTR1 5.11.11

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

ENCLOSURE 4

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT UNITS 2 AND 3
DOCKET NOS. 50-260, AND 50-296

REPLY TO RESULTS OF ACCEPTANCE REVIEW AND REQUEST FOR ADDITIONAL INFORMATION FOR BFN UNITS 2 AND 3 EXTENDED POWER UPRATE APPLICATION

(Non-Proprietary Version)

See Attached:

- Reply to Results of Acceptance Review And Request for Additional Information For BFN Units 2 and 3 Extended Power Uprate Application

Reply To Results of Acceptance Review and Request for Additional Information for BFN Units 2 And 3 Extended Power Uprate Application

As a result of NRC's review of TVA's June 25, 2005, Browns Ferry Units 2 and 3 Extended Power Uprate (EPU) amendment request, the NRC, by letter dated November 18, 2004, requested that TVA submit additional information to enable the Staff to initiate its detailed technical review of the amendment request. Responses to the questions are provided below.

NRC Request 1.

In general, several areas are identified as being bounded by analyses performed as part of the ELTR-1 and ELTR-2 assessments. However, your application does not provide sufficient information to allow the NRC staff to determine the applicability of the ELTR-1 and ELTR-2 analyses to Browns Ferry Plant (BFN), Units 2 and 3. Specifically, information relating proposed operation to the assumptions, evaluations, reviews, and assessments used in the ELTR analyses were not provided. Examples of these include:

NRC Request 1.a.

In Enclosure 4, the EPU Safety Analysis Report (SAR) items are stated to be dispositioned based on confirmation of consistency between BFN and the generic description provided in ELTR-1 and ELTR-2. However, no details are provided to allow the NRC staff to understand how this BFN to ELTR confirmation was performed. Specifically, what criteria, key parameters, etc., were examined to confirm the consistency? Also, identify and justify all the areas where BFN Units 2 and 3 do not satisfy the ELTR criteria.

TVA Reply 1.a.

As discussed in the foreword of Enclosure 4 of the BFN Units 2 and 3 EPU application (which provided the NEDC-33047P Browns Ferry Units 2 and 3 Safety Analysis Report For Extended Power Uprate, or the "PUSAR"), Item 5, TVA identified where the BFN Units 2 and 3 EPU application was based upon ELTR1 or ELTR2 generic evaluations. The table, "Browns Ferry Units 1, 2 and 3 Comparison of ELTR Generic Evaluations to the PUSAR," also included in the foreword of Enclosure 4 of the BFN Units 2 and 3 EPU application, listed those areas where an ELTR1 or ELTR2 generic evaluation was credited. That table identified the:

- PUSAR section references
- Topic (evaluation topic)
- Corresponding ELTR1 or ELTR2 section references,
- ELTR1 or ELTR2 requirements, assumptions, and/or parameters (and their values) for assessing applicability of the generic evaluations to a specific plant, and
- Corresponding BFN Units 2 and 3 requirements, assumptions, and/or parameter values used to compare BFN to the generic evaluation.

This table was compiled by 1) identifying where ELTR1 or ELTR2 generic evaluations were credited in the BFN Units 2 and 3 EPU application, 2) identifying the requirements, assumptions, and/or parameters (and their values) for assessing applicability of the generic evaluations to a specific plant, and 3) comparing the specific BFN Units 2 and 3 corresponding requirements, assumptions, and/or parameter values to ensure that the generic evaluations were bounding for BFN Units 2 and 3. The BFN comparison demonstrated that the Generic Evaluations credited bound and thus represent a conservative conclusion for BFN Units 2 and 3. The subject table has been revised to make it specific to BFN Units 2 and 3 only (BFN Unit 1 information which was provided in the June 25, 2004, application has been removed).

Browns Ferry Units 2 and 3 Comparison of ELTR Generic Evaluations to the PUSAR				
PUSAR Section	**Topic**	**ELTR 1 / ELTR 2 Section**	**ELTR Parameter(s) / Requirement(s) / Assumption(s)**	**BFN PUSAR Comparison**
1.1, 1.3, Table 1-2	Reactor Thermal – Hydraulic Parameters	ELTR 1, Section 1.0, Table 5.1, Appendix C.2	• 20% Thermal Power Increase. • • 24% Steam Flow Increase. • 1095 psia Operating Dome Pressure. • 556° F Dome Temperature. • 99% to 110% Full Power Core Flow Range.	• 20% Thermal Power Increase from original licensed thermal power (OLTP). (15% Increase from current licensed thermal power (CLTP)) • Approximately 22.94% Steam Flow Increase from OLTP. (16.2% Increase from CLTP) • 1050 psia Operating Dome Pressure. • 550.5° F Dome Temperature. • 99% to 105% Full Power Core Flow Range.

Browns Ferry Units 2 and 3 Comparison of ELTR Generic Evaluations to the PUSAR				
PUSAR Section	**Topic**	**ELTR 1 / ELTR 2 Section**	**ELTR Parameter(s) / Requirement(s) / Assumption(s)**	**BFN PUSAR Comparison**
2.3.1, Figure 2-1	Power / Flow Operating Map	ELTR 1, Appendix C.2.3, Figure 5-1, ELTR 2, Section 3.2	• The upper boundary shall be limited to the uprated power level. • The right side of the operating range shall be the same core flow limit as currently licensed. • The left (lower core flow) side of the operating map will be bounded by the new lower limits provided in Table C-1. (99% for BWR3 and 4.)	• The maximum EPU reactor thermal power (RTP) (Points D, E, & F of PUSAR Figure 2-1) corresponds to 120% of the OLTP. • The maximum core flow shown on PUSAR Figure 2-1 corresponds to the previously analyzed core flow range when rescaled so that EPU RTP is equal to 100% rated. • Point D of PUSAR Figure 2-1 corresponds to 99% core flow at 100% EPU RTP.

Browns Ferry Units 2 and 3 Comparison of ELTR Generic Evaluations to the PUSAR				
PUSAR Section	**Topic**	**ELTR 1 / ELTR 2 Section**	**ELTR Parameter(s) / Requirement(s) / Assumption(s)**	**BFN PUSAR Comparison**
3.7	Main Steam Isolation Valves	ELTR 2, Section 4.7	• 20% Thermal Power Increase. • 24% Steam Flow Increase. • 1095 psia Operating Dome Pressure. • 556° F Dome Temperature.	• 20% Thermal Power Increase from OLTP. (15% Increase from CLTP) • Approximately 22.94% Steam Flow Increase from OLTP. (16.20% Increase from CLTP) • 1050 psia Operating Dome Pressure. • 550.5° F Dome Temperature.
4.2.1	High Pressure Coolant Injection	ELTR 2, Section 4.2	• < 75 psi increase in Reactor operating pressure. • The HPCI hydraulic control modification described in GE SIL No. 480 should be installed.	• 30 psi change performed previously. No change from CLTP conditions. • GE SIL No. 480 Installed.

NRC asked TVA to identify and justify all the areas where BFN Units 2 and 3 does not satisfy the ELTR criteria. BFN has satisfied the criteria of the ELTR but in some cases has conducted the evaluation in an alternate manner specified by the ELTR as described below.

Instrumentation and Control – Enclosure 4 to the License Amendment Request - Section 5

ELTR1 Appendix F, Section F.4.1 provides generic guidelines applicable to instrument setpoints for operation at uprated conditions. This guidance utilizes the GE generic setpoint methodology. However, TVA utilized the NRC approved TVA setpoint methodology (TVA Branch Technical Instruction, EEB-TI-28, Setpoint Calculations, Revision 5, February 25, 2000) to generate the allowable values and (nominal trip) setpoints related to the NSSS analytical limit changes associated with the implementation of EPU. The BFN plant-specific methodology has been approved by the NRC, therefore, this approach is allowed by ELTR1 Appendix F, Section F.4.1.

Testing – Enclosure 4 to the License Amendment Request – Section 10.4, and Enclosure 8 to the License Amendment Request, "Justification for Exception to Large Transient Testing"

BFN does not intend to perform large transient testing involving reactor vessel isolation from high power. The technical justification supporting this approach was provided in the BFN Units 2 and 3 submittal Enclosure 4, Section 10.4 and Enclosure 8, "Justification for Exception to Large Transient Testing". Additional information is provided below under NRC Request 4.

Fuel Design – Enclosure 5 to the License Amendment Request – "Framatome Advanced Nuclear Power (ANP) Browns Ferry Units 2 and 3 Safety Analysis Report For Extended Power Uprate"

BFN intends to implement EPU on Units 2 and 3 using Framatome ANP, Inc. (FANP ATRIUM-10) fuel. In support of EPU, Framatome performed a review of the GE EPU fuel and plant related evaluations and analyses to identify those that were specifically fuel-related. The specific fuel related EPU transient and accident analyses were performed for the ATRIUM-10 fuel design as well as additional fuel-related analyses applicable to the uprated condition. These analyses results are reported in Enclosure 5 whose content is consistent with ELTR requirements. This is in keeping with the November 21, 2002, (Reference 1) letter from TVA to the NRC, "Units 2 and 3 – Options for Extended Power Uprate (EPU) and Fuel Vendor Change" for introducing the fuel vendor change and the January 17, 2003, (Reference 2) letter from NRC to TVA, "Fuel Vendor Change, Extended Power Uprate and Maximum Extended Load Line Limit Analyses Plus Submittals" recognizing the transition submittal characterization.

NRC Request 1.b.

TVA has referred exclusively to ELTR-1 and ELTR-2 as the applicable licensing basis for BFN Units 2 and 3. Since the ELTRs do not provide the plant-specific licensing and design criteria, provide a revised enclosure to reflect the appropriate plant-specific licensing and design criteria.

TVA Reply 1.b.

In Enclosures 12 and 13 of its June 25, 2004 application, TVA provided markups of the Extended Power Uprate Areas of Review Matrix and the Extended Power Template Safety Evaluation contained in RS-001, "Review Standard for Extended Power Uprates." Because BFN is not required to meet the 10 CFR 50, Appendix A, General Design Criteria (GDC), TVA marked up the safety evaluation template and areas of review matrix contained in RS-001 consistent with the BFN design bases. However, based on NRC's review of our RS-001 markups and subsequent telephone conferences, we have revised the RS-001 matrices markups to be similar to those submitted by Vermont Yankee for their EPU project.

BFN, like Vermont Yankee, is a pre-GDC plant. The design basis of each BFN unit was evaluated against each of the nine groups of the proposed criteria. Based on the understanding of the intent of the proposed criteria it was concluded that each unit conformed to the intent of the Atomic Energy Commission GDC. With some subsequent exceptions, the current licensing basis is the 70 GDC for Nuclear Power Plant Construction Permits (hereafter referred to as the draft GDC). BFN's conformance to the draft GDC is described in Appendix A of the BFN Updated Final Safety Analysis Report. The major difference between the draft GDC and the final version of the GDC is a consolidation of the criterion from 70 to 64. In general, the basic content of the design criteria are consistent between the draft GDC and the final version.

Enclosure 2 to this letter is the Extended Power Uprate RS-001 Revised Template Safety Evaluation. To aid the staff in preparing the BFN-specific EPU Safety Evaluation, BFN has replaced the numeric values of the GDC in the revised template safety evaluation to incorporate the draft GDC that corresponds to the Browns Ferry criteria that Browns Ferry was reviewed against during each unit's original licensing effort. These changes to the template are identified by change bars in the left margin.

Enclosure 3 to this letter is a Extended Power Uprate RS-001 Revised Areas of Review matrix. The matrix cross references the criteria in the NRC review standard with the information in the BFN Power Uprate Safety Analysis Report, the Framatome Uprate Safety Analysis Report, the draft GDC and the approved ELTR for EPU. Notes have been added to the matrices to provide additional guidance to direct the reviewer to the specific safety analyses and conclusions.

NRC Request 1.c.

Enclosure 4, Section 7.4.1, indicates that the feedwater heater analysis has not been completed. Please provide the completed analysis in the EPU submittal.

TVA Reply 1.c.

Analysis has been completed satisfactorily for EPU conditions on the Feedwater System heaters. The system analysis determined that the tube side of the Number 3 heaters will be subject to increased pressure if the valves downstream of the heaters isolate with the condensate booster pumps in service. This is a result of the higher capacity condensate booster pumps being installed as part of the EPU upgrades.

The Feedwater heaters were analyzed for this increase in pressure and found to be acceptable. The analysis considered an increase in tube side pressure from 450 psig to 525 psig at an operating temperature of 320° F with shell side conditions of both 75 psig and an absolute vacuum at a steam temperature of 650° F. These conditions bound the EPU operating condition for Browns Ferry with the higher capacity condensate booster pumps and EPU steam side conditions.

TVA's review was based upon the current American Society of Mechanical Engineers (ASME) Section VIII Division 1 and Section II codes for all materials except the tube sheets. The review of the tube sheet material was based on the Tubular Exchanger Manufacturers Association tube sheet thickness calculations included in the Heat Exchange Institute Standards for Closed Feedwater Heaters Sixth Edition and the current ASME Section II code. The analysis used a corrosion allowance of 0.080 inch for the carbon steel components. The thickness of the stainless steel tube was determined as a minimum allowable and compared to the existing tube thicknesses. The tubes, tube sheet, channel plate, channel head, channel nozzles, channel openings and heater supports were found to be acceptable without modifications. The manway and partition plate were found to require reinforcing. The upgrades to the feedwater heaters were previously identified in Enclosure 7 to the license amendment application. These will be performed prior to EPU operation to assure acceptable performance with the new system operating conditions.

NRC Request 2.

Items (e.g., in Section 2) of the EPU SAR are dispositioned based on experience and are stated to be confirmed because they will be evaluated for the uprated core prior to EPU implementation. However, these evaluations will be performed close to the reload outage and will only be available in the Supplemental Reload Licensing Report and the Core Operating Limits Report. There is no discussion as to how these confirmations, prior to EPU implementation, will be verified in accordance with the ELTR Safety Evaluation Report, licensee expectations or restrictions, and applicable Title 10, *Code of Federal Regulations* (10 CFR) Part 50, Appendix B requirements.

TVA Reply 2.

The BWR core design and licensing process recognizes at the outset that it is not reasonable to develop a "bounding" core design; that is, "bounding" for all future cycles. Instead, the reload process ensures that fuel design and licensing limits are maintained for reload cores by performing cycle-specific calculations. This is accomplished by performing the required analyses using NRC-approved methods as described in the Core Operating Limits Report (COLR) references listed in Section 5.6.5.b of the plant Technical Specifications. This process ensures that any follow-on core designs will continue to meet all regulatory requirements.

While the equilibrium core may not be bounding, the current regulatory process regarding cycle specific analysis verifies that all licensing limits and regulatory requirements are met. This process is explicitly recognized in ELTR1 and ELTR2 and endorsed in the associated NRC SER. ELTR1, Section 4.2 discusses that the power uprate process of engineering evaluation considers plant operation from a representative fuel cycle viewpoint. Reload analyses are typically performed just prior to a refuel outage utilizing cycle specific information.

The requirements and expectations for engineering and safety evaluations for a power uprate submittal are summarized in ELTR1 Section 4.2, as stated below:

> "The engineering and safety evaluation covers a detailed analysis and assessment of affected aspects of the plant at the selected power level. Plant operation is evaluated from a representative fuel cycle viewpoint, similar to the original SAR analyses. The evaluation effort identifies hardware modifications required to achieve the uprated power.
>
> A Licensing Report, which contains the summary and conclusions of the engineering evaluations, is generated, using the generic outline given in Appendix A, to address safety aspects of operating at the uprated power conditions and planned operating strategy selected from the feasibility phase.

The Licensing Report accompanies the Licensee's application for an increase in the authorized power level, along with any revisions to the Technical Specifications. In its final form, this set of documents will target a specific fuel cycle in which uprated operation is planned. Cycle specific operating limits and evaluations of limiting events will usually be provided separately, similar to current reload analysis and documentation practice."

Additionally, as stated in ELTR1, Appendix E, Section E.2.2 Justification (1):

"The reload evaluation for the first cycle that will implement part or all of power uprate will also provide more specific analysis of these cases for conditions to be experienced for that cycle, including all the exposure history of the core up to that time."

In these statements, the NRC accepted ELTR1 explicitly states that the reload evaluation process rather than the power uprate licensing report is used to confirm that individual core designs (which would include transitional cores) meet regulatory requirements. ELTR1 does not require that the reload analysis be provided as part of the power uprate submittal but instead be provided separately using the current reload analysis and documentation practice (ELTR1 Section 4.2).
The SER for ELTR2 also includes statements that confirm this approach. For example, SER Section 3.4 (SLMCPR) states that the:

"This operating limit MCPR will be documented in each plant-specific power uprate submittal and confirmed for each cycle of operation in the cycle-specific reload analysis."

The SER for ELTR2 also indicates that the NRC recognized that the actual confirmation of cycle-specific core related limits are performed using approved methodologies as part of the reload analysis process. The NRC Safety Evaluation Report for ELTR2 Section 5.1 states:

"The fuel operating limits, such as MAPLHGR and OLMCPR will still be met at the uprated power level. The plant-specific submittal will confirm the acceptability of these operating limits as determined for uprated power conditions. Reload analyses will continue to meet acceptable NRC criteria as specified in GESTAR."

The NRC Safety Evaluation Report for ELTR2 states in Section 5.3 regarding transient evaluations,

"This operating limit MCPR will be documented in each plant-specific uprate submittal and confirmed for each cycle of operation in the cycle-specific reload analysis."

These statements taken together show that the NRC recognizes that the power uprate submittal will be based on a representative core and that the reload analysis process is used to ensure that the limits for the actual operating cycles will continue to be met. This approach is consistent with the approach planned for BFN extended power uprate.

This is further discussed in a November 21, 2002 (Reference 1), letter which provided two options for implementation of EPU concurrent with a fuel vendor change: 1) Use the Constant Pressure Power Uprate License Topical Report Process, or 2) use Extended Power Uprate License Topical Report (ELTR) Process. TVA chose to use the ELTR process which allows the use of a representative core. NRC endorsed TVA's plan in a January 17, 2003 (Reference 2) letter.

The majority of the analyses and evaluations specified in the ELTR to justify operation at EPU conditions are not sensitive to a specific core design. As specified in Section 1.2.3(a) of Enclosure 4 to the license amendment request,

> "Specific analyses required for EPU have been performed for a representative fuel cycle with the reactor core operating at EPU conditions."

This is also addressed in Section 1.2.3 of Enclosure 5 to the license amendment request, which states,

> "A representative ATRIUM-10 equilibrium fuel cycle operating at EPU conditions was used as the basis for the EPU analyses."

The use of a representative core design provided by the ELTR process is consistent with a number of previous extended power uprate submittals (including Hatch, Clinton, and Brunswick).

Analyses that are sensitive to specific core designs are primarily those used in establishing core thermal limits, especially the CPR related limits. These analyses are performed each cycle on a reload basis using NRC approved methods.

As discussed in the NRC SER on ELTR2, the reload process is used to ensure that core designs, including the initial EPU core, will meet all fuel design and licensing limits. This will be accomplished using the NRC-approved methodology listed in Technical Specification Section 5.6.5.b.

A reload licensing analysis is documented in a Reload Licensing Report. This report serves as the primary input document in the generation of the cycle specific COLR, which is required by plant Technical Specifications. Like all potential changes to the plant configuration, each reload is reviewed in accordance with 10 CFR 50.59 requirements. The purpose of this review is to ensure that the cycle specific reload analyses meet all fuel design and licensing requirements, which are identified in the plant Updated Final Safety Analysis Report (UFSAR) and Technical Specifications (including the COLR methodology references in T/S 5.6.5.b). The

10 CFR 50.59 review is required to determine whether prior NRC approval is required for the core design. If it is determined that some portion of the analysis requires NRC approval, then the design is either changed or a NRC submittal is made to obtain approval for the change. An example of a typical cycle-specific submittal is to accommodate a change in SLMCPR due to the core design. For BFN, the reload licensing report for the current cycle becomes part of the plant UFSAR as a unit specific section of Appendix N.

The computer codes used in the EPU core design analyses are discussed in Section 1.2.2 of the Enclosures 4 and 5 to the license amendment request Table 1-3 of both enclosures identifies the codes used as well as their individual approval status. As stated in Enclosures 4 and 5 Section 1.2.2, the application of the codes to the EPU analyses complies with the limitations, restrictions, and conditions specified in the approving Safety Evaluation Report (SERs), as applicable for each code. Any exceptions to these conditions are listed in the respective Table 1-3. The codes used to perform the reload licensing analyses are the same as those provided in Table 1-3 of Enclosures 4 and 5.

The confirmation that the plant will meet the applicable regulations for EPU implementation will be accomplished in accordance with the ELTR SE Report using the established reload processes. The 10 CFR 50.59 review ensures that all design and licensing requirements are met, approved methodologies are used, and determines whether NRC approval is required for each core reload.

NRC Request 3.

In an attached document to Enclosure 5, Framatome Updated Analyses Report (FUSAR), entitled, Licensing Approach for Use of Framatome Fuels, it is stated that:

> . . . the remaining GE14 fuel in the Unit 2 core will be a relatively small batch of twice-burnt fuel (at BOC [beginning of cycle]) located primarily on or near the periphery.

There is insufficient information to establish whether GE14 fuel will be put in critical positions or will be limiting. Since, BFN Unit 2 will be operating with a mixed core, additional information will be needed, such as a mixed core analyses report and a fuel transition report. Also, as BFN Unit 3, will be the first uprated unit using a full core of ATRIUM-10 fuel, additional information, such as the assumptions, limitations, restrictions in the models, and the applications of the models, will be required to establish whether the evaluation models given in Table 1-3 of FUSAR are valid for EPU application. Further, the TVA has not established that the use of the reference equilibrium core will be bounding for the first cycle of EPU operation. Consistent with the guidance provided in Mr. Ledyard B. Marsh's letter to GE dated June 25, 2003, specific operating cycle information must be submitted to show, prior to any approval, to show compliance with all regulations for the proposed transition core design.

TVA Reply 3.

The initial Unit 2 EPU core is assumed to be Cycle 15 with a planned startup in the Spring of 2007. At that time, two ATRIUM-10 reloads will have been loaded into Unit 2 and the remaining GE14 fuel would be twice-burnt fuel at the beginning of the Cycle. Current fuel cycle projections show that the expected number of remaining GE14 fuel assemblies in the initial Unit 2 EPU core to be approximately 104 (out of a total of 764) with a bundle average exposure range of approximately 25 to 40 GWD/MTU at BOC.

The potential locations of the GE14 fuel within the Cycle 15 core will be dictated by exposure. The highest exposure fuel is always loaded on or near the periphery in order to minimize neutron leakage and in order to place the high exposure fuel in low power locations so that they will not exceed their licensing exposure limits.

The Browns Ferry core has ninety-two (92) peripheral locations, those locations in which the bundle would have either one or two reflector faces. An additional twenty (20) locations exist with a corner of the bundle adjacent to the reflector. There are also a number of other near-peripheral locations (one row in from the periphery) in which the power is so low that assemblies in these locations could not be limiting.

Previously burned fuel at the low end of the expected exposure range could only be operated in certain locations in the interior regions of the reactor and still meet licensing exposure limits at the end of the uprated cycle. The most likely interior location for the previously burned (GE14) fuel would be the center cell which is typically loaded with four (4) twice-burnt assemblies. This condition will not be limiting because the most reactive state for these bundles occurs at BOC since all of the twice-burnt bundles have depleted their Gadolinia inventory (except for inconsequential residual amounts). At BOC the limiting bundles in the core are once-burnt which are typically at or just beyond their peak reactivity point. As the once-burnt bundles increase in exposure, the fresh bundles become more limiting due to Gadolinia burnup.

There is one exception to the above discussion regarding the GE14 twice burnt fuel. A single once-burnt GE14 bundle is planned for reinsertion into the Cycle 15 core. This once-burnt GE14 bundle is being discharged for inspection purposes during the upcoming Spring of 2005 refueling outage for Unit 2. In order to maintain interior core symmetry and to ensure it remains non-limiting this bundle will be re-inserted in a near-peripheral location.

To further illustrate these points,GE14 to ATRIUM-10 comparisons of RPF, LHGR, and MFLCPR are given in the following figures. These data are from a Cycle 15 EPU mixed core design from a multi-cycle analysis. This design includes a detailed rodded depletion and provides a representation of the amount of margin to be expected between the GE14 and limiting ATRIUM-10 bundles in Cycle 15. These relative power and thermal limit comparisons show that the GE14 bundles will not reasonably approach the core operating limits.

Unit 2 Cycle 15: Fuel Cycle Core
Radial Power Comparison

Unit 2 Cycle 15: Fuel Cycle Core
LHGR Comparason

Unit 2 Cycle 15: Fuel Cycle Core
MFLCPR Comparison

In terms of actual bundle powers, the maximum GE14 bundle power in this Cycle 15 design is approximately 5.7 MWt with a radial peaking factor of 1.10. This bundle power is equivalent to a bundle operating at a 1.26 peaking factor for current rated power (105% OLTP) conditions. The corresponding maximum ATRIUM-10 bundle power for this statepoint was 6.7 MWt, approximately 15% higher than the GE14. The peak ATRIUM-10 bundle power throughout the cycle was 7.2 MWt with a corresponding radial peaking factor of 1.40, approximately 20% higher than the GE14 maximum.

Some degraded control rod operation due to interference with the fuel channels due to channel bow has recently been noted in the industry. Channel bow can be driven by fluence gradients or potentially by shadow corrosion. So far, this interference has been limited to 'C' and 'S' lattice cores. Browns Ferry is a 'D' lattice core with the control rods located in a non-symmetric 'wide-wide' gap. This wider gap provides more clearance between the channel and control blade than the 'C' and 'S' lattice designs. This gap is not dependent upon fuel type with both the ATRIUM-10 and GE14 designs having the same basic outer envelope dimensions. Furthermore, Browns Ferry core designs mitigate channel bow by:

- limiting channel use to one bundle lifetime,
- making sure fresh fuel is not loaded with a face or corner on the periphery, and

- by conservatively implementing the recommendations of SIL-320 Supplement 3. (SIL-320 addresses channel bow by controlling the orientation of the channel to the flux gradient in each cycle.)

Within the context of this response, these mitigation actions will continue to ensure that the more highly exposed fuel will not become limiting due to channel bow.

The information provided on the computer codes in Enclosure 5 Table 1-3 to the license amendment request, FUSAR Section 1.2.2 on Framatome ANP methodology has the same format and content as that provided in the same sections of the Enclosure 4 to the license amendment request (PUSAR) for GE methodology. These two sections state that the corresponding analyses in the two separate reports were performed with approved codes and calculational techniques in accordance with limitations, restrictions, and conditions specified in the approving SERs with any exceptions identified in the corresponding Table 1-3.

It should be noted that EPU does not challenge vendor fuel analysis methods. EPU does not require the modification of any fuel design parameters but instead is accomplished by radial power flattening. The peak bundle power will remain approximately the same, however; the average bundle power in the core will increase. For example, as noted in Section 2 of the Enclosures 4 and 5 to the license amendment request, the average power density of the fuel in Browns Ferry will increase to 5.17 MW/bundle but this value remains within the range of other operating BWRs. Since no fuel design parameters change with EPU, core designs must still meet the fuel thermal design constraints applied before power uprate. Thus, the same level of information (assumptions, limitations, restrictions in the models, and the applications of the models) has been provided for the A-10 fuel as for the GE fuel.

This question is not applicable for Unit 3 since it will start its first EPU cycle with a full core of ATRIUM-10 fuel and thus will not have a fuel vendor transition core during EPU operation. However, as shown in the previous discussion, the co-resident GE14 fuel in the Unit 2 Cycle 15 transition core will be primarily twice-burnt and will be non-limiting. Therefore, TVA believes the reference ATRIUM-10 equilibrium core is applicable to both units and therefore, includes Unit 3 applicability in this response.

The majority of the analyses and evaluations specified in the ELTR to justify operation at EPU conditions are not sensitive to a specific core design. However, as noted by the NRC, a reference equilibrium ATRIUM-10 core was considered in the submittal for those analyses required by the ELTR that are sensitive to core design (primarily those transients listed in Table E-1 of the ELTR). The use of an equilibrium core design for these analyses has been previously discussed with the NRC. This is consistent with the agreement reached between TVA and NRC as shown in letters dated November 21, 2002, and January 17, 2003.

The terminology "bounding core" is misleading and potentially open to interpretation since it is not reasonably possible to design a core that pushes all fuel design and licensing parameters to their highest acceptable values simultaneously. This is one of the primary reasons that reload analyses are performed on a cycle specific basis for BWRs. No single core can be expected to bound all future cores in regard to the calculated thermal margins required to protect the fuel design and licensing parameters. However, it can also be said that core reload licensing analyses in general are composed of a series of "bounding fuel-dependent analyses" as requested in the NRC letter (Mr. Marsh, NRC to Mr. White, GENE). For example, a bounding cycle-specific SLMCPR analysis pushes the core to the point where the regulatory requirement of <0.1% of the rods experience boiling transition is just met, with appropriate uncertainties. The limiting plant transients are then performed, again with the appropriate conservatisms and assumed limiting failures, to further protect this licensing limit. The basic assumption made is that the core is operated in such a manner that if the limiting transients were to occur at the worst possible time, the individual fuel design parameters and licensing limits are not exceeded. The reload analyses provide the necessary margins to ensure this scenario.

The NRC approved ELTR1 recognizes this and within Section E focuses on the performance of transients for the purpose of confirming that the power uprate will not impact the limiting transients to be performed in the cycle-specific reload process. Specifically, as noted in ELTR1 Section E.2.2:

> "The expanded (over GESTAR) list of transients provided in Table E-1 is intended to confirm that the existing set of reload analysis transients remains valid, and evaluate operational aspects of the power uprate."

Also within Section E.2.2 – as part of justification (1):

> "The reload evaluation for the first cycle that will implement part or all of power uprate will also provide more specific analysis of these cases for conditions to be experienced for that cycle, including all the exposure history of the core up to that time."

In these statements, the ELTR1 explicitly recognizes that the reload analysis process is used to confirm that cycle specific core designs meet regulatory requirements. The justification statement does not require that the reload analysis be provided as part of the submittal, but instead focuses on ensuring that EPU does not cause an impact that would affect the reload process and ensuring that operation of a core at EPU conditions is feasible.

As stated above, the purpose of ELTR1 Section E evaluation is twofold:

3) confirmation that the limiting reload transients remain valid for EPU conditions, and

4) evaluation of operational aspects of the power uprate.

The reference to GESTAR in ELTR1 Section E.2.2 is not meant to limit applicability to GE designed cores. Section E.2 states:

> "In some cases, the transient analysis may not be done by GE. For those plants, equivalent approved methodology will be used and documented in the plant-specific submittal."

The FUSAR (Enclosure 5 of the Units 2 and 3 EPU Submittal) meets all of these expectations and requirements.

NRC Request 4.

Enclosure 8 takes exception to performing any large scale transient testing. The staff does not review the computer codes that are used for balance-of-plant performance and must rely on the startup test program to confirm that the required modifications and EPU analyses have been completed properly and in particular, large scale transient testing is relied upon to demonstrate that the integrated plant performance is properly bounded by the analyses that have been completed. Consequently, the EPU submittal must be revised to identify and describe tests that will be performed that are sufficiently comprehensive to confirm that: a) all plant modifications have been evaluated and implemented properly, and b) integrated plant performance and transient operation is consistent with the analyses that have been completed. Any exceptions based on plant or industry operating experience must describe the experience in sufficient detail to establish the relevance and applicability to the BFN Units 2 and 3 proposed uprate conditions.

TVA Reply 4.

In a February 17, 2005 teleconference, the NRC Staff informed TVA that additional information would be required to support the justification for exception to large transient testing addressed in Enclosure 8 of the license amendment application, and below in this reply. TVA will provide this additional information by April 11, 2005.

In Enclosure 8 of the BFN EPU license amendment request, TVA took exception to the requirement contained in Sections 5.11.9 and L.2.4 of ELTR1 regarding the performance of the two large-scale transient tests, specifically the MSIV Closure Test and the Generator Load Rejection Test. It is TVA's position that initiation of such transients is not prudent. TVA does not plan to intentionally initiate such transients when other testing planned and experience obtained adequately confirm expected integrated systems behavior and response.

TVA modifies the plant design when necessary or appropriate, as authorized under 10 CFR 50.59. As part of the BFN plant modification control process, TVA reviews all Design Change Notices (DCNs) to identify, specify, and document appropriate post-modification testing requirements. The purpose of post-modification testing is to demonstrate conformance with the design requirements of the installed or modified component(s), confirm expected system response, and verify that no undesirable effects were created.

DCNs are prepared and controlled in accordance with TVA procedure SPP-9.3, "Plant Modifications and Engineering Change Control," which require that design engineers identify and document any required verification and/or special testing requirements. Per TVA Procedure SPP-8.3, "Post Modification Testing," engineers review the DCNs to confirm that all necessary post modification testing has been specified and are responsible for final test approval. These programmatic controls ensure, for the design changes, that testing necessary to ensure system performance requirements are met and expected system response is confirmed with acceptable results prior to turnover of the system for operation. These controls also ensure that modifications comply with the requirements of 10 CFR 50 Appendix B. A description of the acceptance testing planned for these modifications is provided below.

Also, in Enclosure 8 to the June 25, 2004 license amendment request, TVA provided details surrounding an unplanned Unit 2 Generator Load Rejection from approximately 3456 MWt. Since then, Unit 3 experienced a full power Generator Load Reject. This was reported to NRC in LER 50-296/2004-002-00 on January 24, 2005 (Reference 5). A summary is provided below.

On November 23, 2004, while Unit 3 was in steady operation at 100% power (approximately 3458 megawatts thermal), a main turbine trip and subsequent reactor scram occurred. A lightning strike occurred on the TVA 500-kV system and a Unit 3 main turbine trip occurred when the rate of speed change measured by the Electric Hydraulic Control (EHC) System exceeded the maximum rate anticipated by the turbine logic system. Consequently, as required by system design, the turbine tripped and a subsequent reactor scram occurred. All expected system responses occurred.

The BFN Unit 2 EHC logic is configured identically to that of Unit 3 however, during the transient, measured Unit 2 speed change did not exceed limits. Unit 2 continued power operation.

Actuation of primary containment isolation system occurred due to the expected temporary lowering of reactor water level below the actuation setpoint. This logic isolates Shutdown Cooling, the Reactor Water Cleanup (RWCU) System, and normal reactor building ventilation. The logic also initiates both the SGT System, the CREV System, and retracts Traversing Incore Probes (if extended). The normal heat rejection path remained in service throughout the event. Reactor water level was recovered to the normal operating range by the normal reactor water level control system. Neither the High Pressure Coolant Injection (HPCI) nor Reactor Core Isolation Cooling (RCIC) systems were used during this event. Reactor water level did not drop to the auto-initiation point for these systems, and they were not manually placed in service. No safety-relief valve (SRV) operation occurred during the trip transient. The post-trip review confirmed that peak reactor pressures remained below the nominal SRV lift setpoints.

UFSAR Sections 14.5.2.4 and 14.5.2.5 specifically address the main turbine trip event. Main turbine bypass valves are assumed to function in the discussion under Section 14.5.2.4. Section 14.5.2.5, however, assumes that the main turbine bypass valves do not function and therefore is the more severe event. This analysis assumes the most limiting conditions of: end of cycle fuel exposure conditions, a core power of 100% of rated and normal feedwater temperature. The analysis indicates that no safety limits are exceeded for such a transient scenario. The actual plant conditions for this event were less limiting than those described in the UFSAR section 14.5.2.5 analysis, and the Unit 3 event is fully bounded by this analysis.

Browns Ferry Units 2 and 3 EPU Planned Modifications		
Modification	**Description**	**Testing**
Main Turbine	• Replace HP Turbine diaphragms and buckets. • Replace springs, bonnets, washers, bellows, & bolting on 6 cross around relief valves to permit increased set pressure. • Replace miter bend elbows in the condenser spray piping with long radius elbows to reduce back pressure.	Turbine Balancing (if required) Overspeed Test Control and Stop Valve testing Relief valve bench testing
Turbine Sealing Steam	• Modify the size of the steam seal unloader valves and associated piping to allow the turbine sealing system to accommodate the larger steam flow requirements.	Condenser Vacuum testing monitor steam seal header pressure. Calibration of the Steam Seal Header Pressure controller. Inservice leak test
Condensate Pumps	• Replace 2 impellers in each of 3 pumps. • Install 3 - 1250 hp motors. • Recalibrate relay settings. • Recalibrate/replace pump & motor instrumentation. • Modify HVAC ductwork.	Verification of pump flow and head. Monitoring of pump and motor parameters (flow pressure, temperatures, etc.). Instrumentation calibration and functional testing.
Condensate Booster Pumps	• Replace 3 pumps. • Install 3 – 3000 hp motors. • Recalibrate relay settings. • Recalibrate/replace pump & motor instrumentation. • Modify HVAC ductwork.	Verification of pump flow and head. Monitoring of pump and motor parameters (flow pressure, temperatures, etc.) Instrumentation calibration and functional testing.

Browns Ferry Units 2 and 3 EPU Planned Modifications		
Modification	**Description**	**Testing**
Steam Packing Exhauster Bypass	• Install 24" piping & flow control valve to accommodate increased condensate flows at EPU conditions.	Valve testing
Condensate Demineralizers	• Install 1 new vessel with valves & digital controls. • Upgrade controls on 9 existing vessels to digital. (Unit 3 only) • Replace valves for increased reliability.	Control system functional testing. Initial installation Startup test (flow, temperature, pressure, etc.)
Main Condenser Extraction Steam Bellows	• Replace #2, #3 and #4 bellows with upgraded bellows.	Installation Examination
Feedwater Pumps and Turbines	• Replace 3 pumps. • Recalibrate pump instrumentation and control system for increased flows at EPU conditions. • Replace turbine/pump coupling. • Replace turbine rotor, diaphragms and buckets. • Recalibrate/replace turbine instrumentation.	Balancing Overspeed testing Controls Tuning Verification of pump flow and head. Monitoring of pump and turbine parameters (flow pressure, temperatures, etc.) Instrumentation calibration and functional testing.
Feedwater Heaters	• Upgrade heater shell pressure certification. • Replace level transmitters on FWHs 1, 2 & 3. • Repair / replace 18 nozzles on FWHs 1, 2 & 3. • Replace relief valves on FWHs 1, 2 & 3. • Relocate extraction steam nozzle & shorten extraction steam line on FWH 3. • Install new impingement plate & steam duct inside FWH 3. • Reinforce / reweld pass partition plates in all FWHs.	Relief Valve bench testing Installation Testing (flow, temperature, pressure, etc.) Instrumentation calibration and functional testing. Inservice leak rest.

Browns Ferry Units 2 and 3 EPU Planned Modifications		
Modification	**Description**	**Testing**
Moisture Separators	• Change vanes and add perforated plate on moisture separators. • Modify internal drains as needed.	Moisture removal effectiveness testing Inservice leak test. Installation testing (flow, temperature pressure, etc.)
Main Generator System	• Recalibrate/replace pressure regulators and pressure switches. • Increase generator hydrogen to 75 psig to operate at increased loads.	Field Installation testing. Instrumentation calibration and functional testing. Monitoring of system (i.e., voltage, amps, temperature) during power ascension.
Main Bank Transformers	• Install 3-500 MVA transformers per unit. • Install 2-500 MVA spares (U1/2 & 3). • Upgrade oil and water deluge systems. • Upgrade relaying as needed. • Replace 5 – breakers & disconnects.	Field Installation testing Deluge spray down testing Performance monitoring
Isolation Phase Bus Duct Cooling	• Modify Isolation Phase Bus Duct Cooling System to remove Bus Duct heat under EPU conditions.	Verification of system flow, both air and water.
EHC Software	• New program inputs & logic for EPU conditions.	Verification of control functions Turbine Valve setup Controls Tuning
Technical Specification Instrument Respan	• Setpoint/scale changes on multiple instruments.	Calibration per applicable Surveillance Requirements instructions.
Balance Of Plant Instrument Respan	• Replace/recalibrate multiple instruments.	Calibration per applicable procedures.
Drywell Building Steel	• Modify building steel beams and connections as required for load changes at EPU conditions.	Applicable structural installation testing

Browns Ferry Units 2 and 3 EPU Planned Modifications		
Modification	**Description**	**Testing**
Main Steam Supports	• Modify supports as required for load changes due to EPU conditions.	Applicable structural installation testing Vibration monitoring of Main Steam and Feedwater Piping and components
Torus Attached Piping	• Modify supports and snubbers as required due to EPU conditions.	Applicable structural installation testing
Main Steam Isolation Valves	• Replace MSIV poppets and modify operators as required to reduce differential pressure across MSIVs at EPU conditions. • Install 2-inch MSIV stems as required due to increased stem forces caused by EPU MS flow increase.	Stroke time testing Applicable Technical Specifications testing Performance monitoring
Reactor Recirculation Pump Motors	• Revise electrical protection system setpoints. • Revise temperature monitoring setpoints. • Assess additional heat load on plant HVAC & cooling water systems. • Assess power cable voltage drop increase due to higher current. • Revise pump/motor vibration monitoring setpoints. • Re-rate pumps and motors for 120% Power/105% Core Flow operating conditions.	Applicable instrumentation calibrations. Vibration monitoring Controls tuning and system operation during vessel hydro
Jet Pumps	• Install sensing line clamps to reduce jet pump vibration due to vane passing frequency at Recirc pump speeds.	None required
Local Power Range Monitors	• Replace 5 LPRMs with those qualified for EPU conditions.	Applicable Technical Specifications testing and calibration

Browns Ferry Units 2 and 3 EPU Planned Modifications		
Modification	**Description**	**Testing**
ICS/SPDS	• Update as needed based on NSSS and BOP instrument changes.	Performance monitoring
Main Steam Relief Valves	• Upgrade pressure actuation logic to Safety Related	Bench testing Cycling at power
Steam Dryer	• Perform identified modifications required to maintain Dryer structural integrity at EPU Conditions.	Testing is discussed in TVA's Reply 5.a (1).
Vibration Monitoring	• Install temporary sensors based on ongoing analyses. • Conduct testing program during power ascension.	Collect and analyze vibration data on selected systems

NRC Request 5.

The NRC staff noted that in several review areas there was insufficient information provided to arrive at an adequate safety conclusion, as described in the template.

NRC Request 5.a.

The following issues were identified with TVA's analysis provided in Enclosure 9 (GE-NE-0000-0023-1250-1) of the submittal supporting the structural integrity of the BFN steam dryer under EPU conditions.

TVA Reply 5.a

Background

TVA's responses to the specific questions concerning the steam dryer analysis submitted in Enclosure 9 of the BFN Units 2 and 3 EPU application is provided below. As stated in the responses below, TVA continues to collaborate with the industry to develop improved steam dryer load definitions and improve the steam dryer analytical model. However, there is additional basis to support a conclusion that the BFN Units 2 and 3 steam dryers will maintain their structural integrity under EPU flow conditions. This information is presented below.

The steam dryer failures occurring at the Quad Cities and Dresden power plants are in part related to higher steam flow velocities causing changes in Steam Dryer component loadings and consequent failure due to fatigue. Experience has also shown that the BWR3 "square hood" design of these plants appears to be more susceptible to the phenomena than the BWR4 "slanted hood" design installed in the BFN plants. Plants of BWR4 design currently operating at greater than 110% of OLTP, have not experienced the failures exhibited in the BWR3 fleet dryers as confirmed through moisture content monitoring and visual inspections. It should be noted that Brunswick 1 is currently operating at 120% power with no apparent structural problems.

TVA has evaluated BFN current and forecast EPU steam flow velocities against the existing operating experience for BWRs that have implemented EPU. As shown in the following table, the evaluation shows that forecast main steam flow velocities (calculated at the reactor vessel steam nozzle) are substantially less than those measured at the Quad Cities and Dresden plants that have experienced steam dryer component failures.

Plant	Design	Hood Design	Uprate Power Level (% of OLTP)	OLTP MSL Velocity[1] (fps)	EPU MSL Velocity[1] (max fps)	Comments
Dresden 2, 3	BWR3	Square	117%	168	207	202 fps average MSL velocity
Quad Cities 1, 2	BWR3	Square	117%	168	226	202 fps average MSL velocity
Vermont Yankee	BWR3	Square	120%	140	168	No pressure increase
Brunswick 1	BWR4	Slanted	120%	129	149	Unit 1 dryer inspection after one cycle of 113% OLTP revealed no cover plate/hood fatigue failures.
Brunswick 2	BWR4	Slanted	115%	129	149[2]	
Hatch 1	BWR4	Slanted	115%	119	134	No cover plate/hood fatigue failures.
Hatch 2	BWR4	Slanted	115%	121	140	
Browns Ferry 1,2,3	BWR4	Slanted	120%	128	153 (forecast)	Units 2 and 3 currently operating at 105% of OLTP.

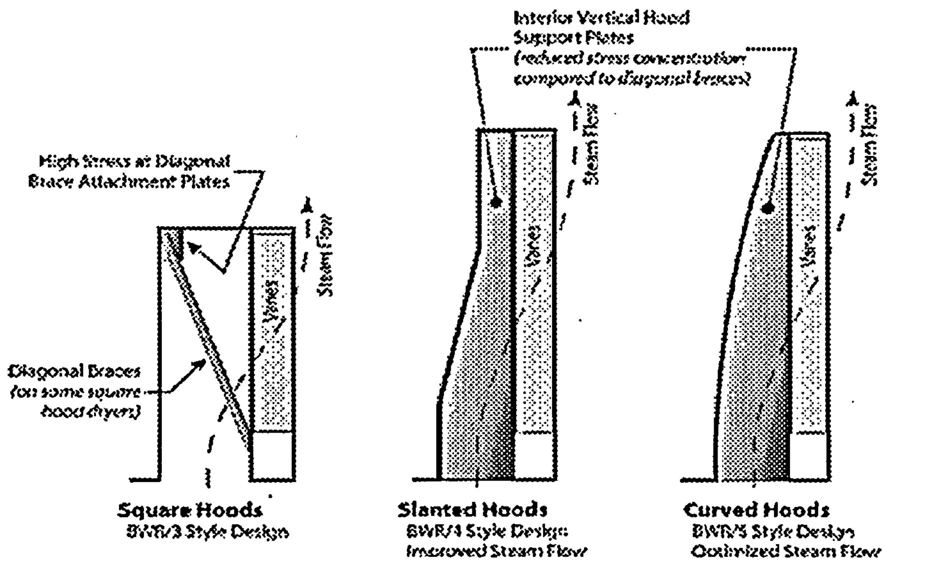
1.Main Steam Line (MSL) velocities at the reactor vessel nozzles in feet per second (fps).
2.Predicted MSL velocities at 120% of OLTP.

The Main Steam line flow velocities attributed to contributing to dryer failures at Quad Cities and Dresden Nuclear Plants are significantly higher than the Main Steam line flow velocities at BFN. In fact, the predicted BFN flow velocities at EPU conditions are less than the Dresden and Quad Cities flow velocities at the original licensed power level. Data from the table above demonstrates that the Dresden and Quad Cities maximum steam line velocities of 207 fps and 226 fps are respectively 35.3% and 47.7% higher than the forecast EPU steam line velocity for BFN. The BFN EPU forecast steam line velocity of 153 fps is comparable with the current Brunswick 120% EPU steam line velocity of 149 fps and the Hatch Unit 2 115% EPU steam line velocity. Both the Hatch and Brunswick dryers are of the same dryer type as installed in BFN (slanted hood), and have had favorable dryer operating experience at these steam velocities.

The Dresden and Quad Cities plants are BWR3 designs which utilize a steam dryer design with square end hoods, making them more susceptible to adverse loading conditions than the BFN BWR4 design with slanted steam dryer end hoods. The BWR4 hood design incorporates a slanted hood face section and additional dryer bank stiffener plates, which provide significantly more restraint to the hood face over and above the BWR3 design.

Figure 1

SIL 644 Supplement 1
Basic GE Steam Dryer Hood Types



NRC Request 5.a.(1)

The excitation source for flow-induced vibration effects and, thus, the actual applied forcing function on the BFN steam dryer has not been adequately determined.

TVA Reply 5.a.(1)

Identification of the specific excitation source(s) for flow-induced vibration effects is an on-going industry program. Acoustical circuit analyses have been developed to identify the contributions from main steam line components such as the main steam line nozzle, safety/relief valve piping junctions, piping connections such as the HPCI and RCIC system connections, equalizing header connections, and "D" ring connections. To further support the BFN dryer evaluation, TVA plans to instrument BFN Unit 1 to obtain Main Steam Line pressure data for power levels up to the proposed 120% of OLTP. These plant-specific parameters will be utilized to develop an acoustical circuit analysis to identify the pressure loading imposed on the dryer components. This information will be compared to TVA's current analysis of record, which incorporates the GE generic load definition. This approach will incorporate the existing sources for the frequency domains of concern.

The TVA MSL monitoring program will incorporate the use of strain gauges to determine hoop stress and consequently pressure loading. As currently planned, strain gauges will be installed on each MSL in close vicinity to each of the selected MSL nozzles. In addition, strain gauges will be mounted near selected flow elements (circumferentially mounted). Also MSL pressures will be monitored with dynamic pressure transducers located at the MSL venturis instrument lines, EHC pressure sensing header, steam equalizing header, and the reactor water level instrument lines. This information will be utilized to augment and reference the strain gauge information obtained from the MSL piping. The dynamic pressure information will contain plant-specific steam flow induced dynamic pressure fluctuations as a result of the plant-specific flow induced vibration sources from all contributors. These strain gauges, accelerometers and transducers will be installed on Unit 1 as the lead unit for EPU implementation.

In conjunction with this monitoring program, piping vibration will be monitored through the use of accelerometers mounted on the MSL piping. Accelerometer locations include adjacent locations to the strain gauges, HPCI system connection, and selected Relief Valve connections. For BFN, the lead unit to implement EPU, the MSL monitoring program will be performed in order to provide for controlled power ascension and load confirmation using the MSL pressure data, acoustic circuit analysis applied dryer loads and subsequent dryer stress analysis for each of the EPU power ascension test plateaus.

TVA has performed a detailed peer review of the GE Steam Dryer load definition methodology and analysis performed for BFN. This was conducted by TVA's lead discipline Engineering Manager with technical support by industry experts. The review focused on the following areas of the steam dryer analysis:

- Development of loading definition/time histories
- Application of plant-specific response spectra for analysis
- Adequacy of load combinations
- Application of loading inputs to finite element analyses
- Adequacy of finite-element model developed for use in analyses
- Confirmation of adequacy of analyses for all three BFN units

This peer review resulted in the correction of a number of analyses assumptions/processes to be consistent with the required plant-specific analysis and provided TVA with assurance that all phases of the analysis were adequate.

However, to further address current steam dryer issues and concerns, TVA will continue to perform peer reviews, utilizing TVA and industry experts, for the review of refinements in the load definition as well as dryer structural models.

As a result of these efforts TVA will have sufficient tools and information to instrument its lead EPU plant's Main Steam Lines during power ascension to verify that the dryer stresses remain within acceptable limits.

NRC Request 5.a.(2)

Many uncertainties exist in the load definition that attempts to bound the complex nature of the fluid excitation forces acting on the dryer at EPU conditions. Also, the ability to construct a dynamic response spectrum to bound the dryer response is questionable, because its frequency content and magnitude are extrapolated from other reactors pressure measurements in stagnant regions located significantly away from the critical dryer hood surfaces.

TVA Reply 5.a.(2)

Development of improved load definitions to reduce the uncertainties regarding the interaction of fluid excitation forces acting on the dryer is a current industry initiative. In order to expand on the GE generic load definition, which was based on previous dryer pressure loading data obtained from instrumented BWR plants, TVA will utilize BFN specific plant data obtained from its operating plant performance up through 120% OLTP as previously described in TVA Reply 5.a.(1) above. These main steam line operating pressures will be utilized as input in the acoustical circuit analysis specific to BFN. The acoustical circuit analysis will develop pressure loads imposed on the dryer surfaces, and will enable dryer stress analysis.

The BFN Unit 1 MSL pressure data to be obtained through 120% OLTP operation will provide plant-specific input performance data which will also contribute to reducing the uncertainty in the BFN analysis. As stated earlier, during EPU power ascension the Browns Ferry lead EPU unit will be instrumented for MSL pressures. This power ascension information will be collected at each of the EPU power ascension test plateaus and compared against the design inputs to be developed for dryer loads and stresses. This EPU power ascension testing will enable a comprehensive validation that the plant performance is within the design analysis for dryer stress limits.

NRC Request 5.a.(3)

The maximum calculated stress for the unmodified steam dryer at current licensed thermal power (CLTP) conditions is too high and reflects large uncertainty in simplifying the complex nature of loads experienced at EPU conditions.

TVA Reply 5.a.(3)

The maximum calculated stress analyses originally performed for the BFN dryers yielded high stresses as a result of the generic load definition, which applied a maximum uniform load on the dryer. This maximum uniform load was based on the available dryer loads obtained from historical instrumented dryer test programs. TVA's final analysis will utilize its plant-specific performance inputs from multiple MSL locations. Utilization of plant-specific data will reduce the loading uncertainty in the initial analysis and it is expected that the initial analysis will remain bounding.

NRC Request 5.a.(4)

Scaling down the results from the dynamic analysis by a presumed factor on stresses at all locations may be not conservative since the true stress at some locations is undetermined.

TVA Reply 5.a.(4)

The submitted dryer analysis required the scaling down of the dynamic analysis results due to the conservative assumptions embedded into the GE generic load definition that incorporated broadened pressure amplitude peaks and uniform pressure loadings imposed on the dryer surfaces. The scaling factor was derived so that the predicted dryer stresses are consistent with the current condition of the dryer under OLTP operations. To satisfy this condition, a scaling down factor was derived so that the flow induced vibration peak stress of the dryer as determined from the dynamic analysis of the dryer is just under the fatigue endurance stress limit for the OLTP condition. TVA's continued evaluations will incorporate the ability to narrow these pressure peaks through the acoustical circuit analysis and instrumented plant-specific measurements to develop plant-specific steam flow induced dynamic pressure amplitude and distribution on the dryer.

NRC Request 5.a.(5)

The pressure on the faces of the dryer extrapolated from CLTP to EPU has not been validated. No information on pressures above CLTP is available.

TVA Reply 5.a.(5)

Plant-specific MSL pressure loading will be obtained for the BFN Unit 1 up through the proposed licensed power. Multiple power level measurements throughout the power ascension will provide a definitive plant response to the increased steam flow and component and excitation source behavior. Plant response will permit the development of anticipated reaction throughout the EPU power ascension to 120% OLTP. Industry efforts are underway to conduct benchmark validation experiments utilizing the scale model test facility and acoustical circuit analysis in order to demonstrate the predictability of the methodology to replicate measured test performance. Other industry validation efforts will be performed on planned replacement dryers (Quad Cities Unit 1) which will be instrumented. The information obtained from these industry initiatives will be integrated into the TVA analysis program to further validate analysis methodologies.

The BFN steam dryer power ascension monitoring will limit power increases so that dryer stress as analyzed are maintained within allowable limits. If power ascension testing reveals greater main steam line loading than anticipated, further acoustical and structural analysis will be performed to insure dryer stresses remain within allowable limits.

NRC Request 5.a.(6)

The formulation used to define the plant-specific load at BFN has not been benchmarked against test data.

TVA Reply 5.a.(6)

Benchmarking of the acoustical circuit analysis is in process against the scale model test facility. This is necessary to validate the analysis methodology. Power response trending from industry experience will be verified to support BFN analysis basis. Benchmarking of plant-specific loads trending will be performed against an instrumented replacement dryer for Quad Cities Unit 1. Information obtained from these industry efforts will be integrated into TVA's dryer analysis program as required.

TVA plant-specific test data will be obtained through the design validation testing program to be implemented for EPU and compared against the acceptance criteria developed through the TVA analysis program.

Summary of TVA Reply 5.a

Based upon the current susceptibility analysis, as well as known industry experience, TVA proposes to modify the BFN steam dryers. These modifications will be similar in scope to those made for the Brunswick BWR4 dryers, currently operating at extended power uprate levels. Lessons learned from recent BWR dryer modifications, such as lower stress welded joint configurations and optimized gusset configurations are being incorporated into the BFN modifications. TVA's proposed modifications currently include:

- Additional reinforcement of the existing cover plate 1/4 inch fillet weld to attain a full 3/8 inch fillet weld.
- A gusset on each dryer hood face is planned that will provide additional reinforcement to the cover plate and hood face. The height of the added gusset will be optimized based on the most mitigating configuration for reducing overall dryer stresses, improving flow straightening contributions and dose reductions for the underwater divers implementing the modifications. Welded attachments to the existing dryer components will invoke the objective of minimum weld stress design approach by increasing the weld area and minimizing weld stress concentrations.
- TVA will also modify the dryer tie-bar configuration to incorporate an improved design. This design will provide both increased load capability for the tie-bars and also introduce greater flexibility between dryer banks and reduce weld stress concentrations at the attachment points.

The MSL monitoring program for BFN Unit 1 will be initiated during plant startup and power ascension. TVA will have the results of analysis and testing available as needed to support staff review.

NRC Request 5.b.

Enclosure 4, Section 10.3.2 discusses Mechanical Environmental Qualification. Specifically identify what equipment will be affected, what non-metallic components are being referred to mechanical equipment, and the basis for acceptance.

TVA Reply 5.b.

Although BFN does not have a licensing requirement to establish or maintain a formal mechanical environmental qualification program, TVA systematically evaluated the effects of operation at EPU conditions on the mechanical components of safety-related equipment. As discussed in Enclosure 4, Section 10.3 of the BFN license application TVA determined that changes in normal and post-accident environmental conditions resulting from operation at EPU conditions would not impact the ability of mechanical equipment to perform its safety-related functions. Based on similarity of the units, there are no differences between Units 1, 2, and 3 with respect to Mechanical Equipment Qualification. A discussion of the assessment performed is provided below.

The mechanical environmental qualification assessment was performed assuming operation at EPU conditions (i.e., 3952 MWt). This assessment evaluated the effects of changes in plant normal and post-accident environmental conditions resulting in operation at EPU conditions on the mechanical components of safety-related equipment. As discussed in Enclosure 4 to the license amendment request, Sections 10.3.1.1 and 10.3.1.2, and tables 10-1 and 10-2 of Enclosure 4 to the license amendment request, the changes in the normal and post-accident environmental conditions are minimal.

The external condition assessments for EPU conditions were based upon the analyzed environments developed for the environmental qualification of safety related electrical equipment. These values were utilized for mechanical equipment assessment.

To demonstrate the ability of safety related mechanical equipment to function, the conditions both internal and external to the equipment were considered. The internal conditions are associated with the process conditions of mechanical equipment and external environmental conditions were associated with for normal, abnormal, and accident conditions.

For environmental qualification of safety related electrical equipment the plant has been categorized into two primary designations relative to environments: mild and harsh. A mild environment is one in which the combination of both the normal and post accident conditions do not represent a threat to the performance of equipment even if the equipment's environmental qualification has not been formally documented. The environmental limits for mild environments are such that mechanical equipment would not be adversely affected by these environmental conditions.

In order to demonstrate the above described qualification for electrical equipment in harsh environments, the plant environments are established for normal, abnormal and accident conditions on an area/room basis. Bounding accident environments are created by loss of coolant accidents (LOCA) and/or high energy line breaks (HELB).

Non-LOCA accidents (i.e., fuel handling accidents or control rod drop accidents) do not produce limiting conditions and, therefore, are not discussed. It has been previously concluded that no areas of the plant that were considered mild prior to extended power uprate will transition to harsh due to the effects of extended power uprate.

Assessment Process

A two step process was followed to assess mechanical safety related equipment located in harsh environments. First, a systematic, area/room, assessment of the changes in the environments due to extended power uprate was performed. Secondly, for equipment in areas whose environments will be changed by EPU, effects of environmental changes on mechanical equipment were assessed. This assessment focused on the effects imposed by the changes created by extended power uprate. This assessment utilized inputs (temperature, pressure, flooding, humidity and radiation) from the normal and accident analyses which provided environmental conditions for each room to be assessed.

Effects of Temperature

Elevated temperature can and will alter the mechanical properties of materials, from both short term and long term exposures. These effects are dependent on the magnitude of temperatures that are reached and the duration for which elevated temperatures are sustained.

Extended power uprate analyses have established temperature profiles for the environments in the plant. In some cases the temperatures remain the same while in other locations the temperatures increase. The categorizations of temperature changes are assessed as follows:

Small Increases in Temperature

The temperature in some area/rooms (e.g., Elevation 565 ft General Floor area, RWCU heat exchanger room) increased on the order of 5° F as a result of extended power uprate. Mechanical device properties and operation will not be adversely affected by an increase in temperature of this magnitude and, therefore, no additional assessments were performed to ensure the ability of mechanical equipment in these areas/rooms to perform safety functions.

Temperature Increases with Short Duration Peaks

The extended power uprate environmental profiles for some rooms (e.g., pressure suppression chamber room, RCIC and Core Spray room) indicate that there are short duration (typically 10 minutes or less) rapid increases and subsequent rapid decreases in temperatures. These short duration excursions in temperatures have been determined to pose no detrimental effect on the operation of mechanical equipment. This is based on the relatively short duration over which they occur, and the time response of the material heat-up relative to the length of time for which the increase is in effect. Non-metallic items are internal to the devices and thus are thermally shielded from the effects of short term temperature excursions. Additionally, insulation covering many of those components provides thermal shielding for the component. For short term temperature transients, typically less than 10 minutes, the capability of mechanical equipment to perform safety related functions will not be adversely affected.

<u>Temperature Increases on Equipment that is not Required</u>

The maximum temperature increase for some rooms (e.g., HPCI room) is caused by a line break for which mechanical equipment in the room is not required to function. In these cases, the temperature increases will have no consequence.

<u>Process Fluid Driven Environmental Conditions</u>

For temperature increases in the environment, even after a line break accident, the maximum temperature in the environment will not approach the temperatures imposed by the process fluids in normal operation. For these cases, no additional effects are imposed by an increase in the environmental temperatures.

Only the drywell remains outside those cases listed above and thus specific components were identified and assessed for effects. The limiting event for the drywell temperature is the main steam line break for which the peak temperature of 336° F did not increase.

The safety related mechanical equipment in the drywell can be segregated into the following categories with regard to temperature;

Category 1 - mechanical devices containing only metallic components,

Category 2 - mechanical devices containing non-metallic components that experience process conditions that are more severe than the post-accident environmental conditions, and

Category 3 - mechanical devices containing non-metallic components for which the normal process conditions are less severe than the post-accident environmental conditions.

For items in the first category, with respect to the metallic components of mechanical devices, mechanical properties would not be affected by small increases in temperature due to EPU in the general temperature range indicated. Therefore, no further review is required for these items. For items in the second category, the devices as well as their non-metallic components experience normal operating process conditions which are more severe than the EPU post-accident environmental conditions. The normal process fluid conditions are controlled by Technical Specifications. These limits are not changed by EPU and therefore, the worst case conditions non-metallic components would experience have not changed. Hence, these items also require no further mechanical assessment. For those remaining mechanical devices in the third category, an additional assessment was performed.

Components in the drywell are categorized and assessed (if required) below.

Control Rod Drives (CRD)

The CRDs are Category 2 items which are located on the under side of the reactor vessel and extend into the reactor core. The components that reside in the core region reach higher temperatures than the temperature profiles for the drywell; thus, the components in that area are designed for that service. The portion of the CRD that mates with the lower portion of the reactor vessel, which contains non-metallic O-rings, is heated to reactor vessel temperatures which are in excess of the environmental temperature profiles. These O-rings are designed for the under vessel service temperature conditions and; therefore, qualified for the EPU post accident environment since it is less harsh than the normal operating environment.

Check/Motor Operated/Manual Valves

Several safety related valves are located in the drywell such as those in the Residual Heat Removal System, Reactor Recirculation System, Reactor Building Closed Cooling Water System, HPCI System, RCIC System, and the Core Spray System. These valves provide reactor coolant pressure boundary and/or primary containment isolation functions. Valves in the CRD system performing primary containment isolation function contain non-metallic components in the seating surfaces. These valves are in the BFN local leak rate testing program and failure of the non-metallic seating surface would be detected. Total failure of the seating surface would allow some minor leakage but would not prevent the performance of the safety function.

Pneumatic Valves

The drywell has two applications of safety related pneumatic valves in Category 3 above: the reactor head vent valves and the Sampling System isolation valve. The reactor head vent valves are normally closed and fail closed. The Sampling System isolation valve is also normally closed and fails closed. Non-metallic components are required to open these valves but cannot impede the closure due to failure. In both cases, the result of a failure in the non-metallic components in the operators (e.g., diaphragm, seals, etc.) would not affect the safe shutdown of the plant since the valves would fail to the safety related position (closed).

Torus to Drywell Vacuum Breakers

The torus to drywell vacuum breakers are designed to control pressures between the torus and the drywell. These are check valves and they contain no non-metallic components (i.e., category 1 above).

The internal process and external environmental temperature changes associated with extended power uprate will have no adverse effect on mechanical equipment's capability to perform safety related functions.

Effects of Radiation

Radiation in excessive amounts can and will alter the mechanical properties of non-metallic materials both in short term and long term exposures. Equipment assessments were based on an increase in normal and post accident doses due to EPU as provided in Enclosure 4 to the license amendment request, Table 10-2.

As part of the transition to 24 month fuel cycles and the associated extended exposures beyond the 1000 EFPD limit of TID 14844, the post accident fission product distribution was determined using the ORIGEN computer code versus the fission product distribution of TID 14844. While the TID 14844 fission product distribution is a simplified approach, the ORIGEN fission product distribution calculation explicitly accounts for individual isotopic distributions by fuel type and exposure. The radiation doses for BFN continued to utilize the TID methodology of released inventory fractions and chemical forms for determining the post accident radiation doses with the ORIGEN fission product distribution. By implementing the ORIGEN fission product distribution, the fission product source term is more accurately defined on an individual isotopic basis. This resulted in some dose constituents changing in opposite directions (e.g., post accident gamma sources contained in piping increase while the airborne gamma doses experience a larger, offsetting decrease). These offsetting changes are due to the transition in fission product distribution from TID 14844 to ORIGEN determination rather than being due to extended power uprate.

For most areas containing safety related mechanical equipment, the increase in normal dose has no significant impact on mechanical devices since the normal dose is significantly less than the total integrated dose. For the remaining areas (i.e., RWCU areas, main steam valve vault, drywell), the effects of the normal dose increase are assessed below:

RWCU and Main Steam Valve Vault

There is no safety related mechanical equipment inside the RWCU areas that contain non-metallic components. There are check valves inside the main steam valve vault that contain non-metallic soft seats. These valves are part of the primary containment pressure boundary and are cycled periodically during reactor shutdowns. The valves are tested during Appendix J leak rate testing which would provide indication of deterioration. Although these seats may deteriorate over time, they are not likely to disintegrate immediately or entirely. A total failure of the seats would allow some minor detectable leakage but will not prevent the systems from performing their safety functions.

Similar to the temperature discussion above, the safety related mechanical equipment in the drywell can be segregated into the following categories with regards to radiation;

Category 1 - mechanical devices containing only metallic components, Category 2 - mechanical devices containing non-metallic components that experience process conditions that are more severe than the post-accident environmental conditions, and

Category 3 - mechanical devices containing non-metallic components for which the normal process conditions are less severe than the post-accident environmental conditions.

For items in the first category, metals are not affected by radiation in the levels experienced in the drywell. Therefore, no further assessment is required for items in this category. For items in the second category, the devices as well as their non-metallic components experience normal operating process conditions which are more severe than the EPU post-accident environmental conditions. The normal process fluid conditions are controlled by Technical Specifications. These limits have not changed by EPU and therefore the worst case conditions which non-metallic would experience have not changed. For those remaining mechanical devices in the third category, additional assessments were performed.

Only the drywell remains outside those cases listed above and thus specific components were identified and assessed for effects.

Control Rod Drives (CRD)

The CRDs are a Category 2 items located on the under side of the reactor vessel and extending into the reactor core. The components that reside in the core region experience higher radiation levels than the general area radiation profiles for the drywell; thus, the components in that area are designed for that service. The portion of the CRD that mates with the lower portion of the reactor vessel, which contains non-metallic O-rings, is also in a radiation field which is in excess of the general area environmental radiation. These O-rings are designed for the under vessel service conditions and; therefore, qualified for the EPU post accident environment since the post accident environment is less harsh than the normal operating environment.

Check/Motor Operated/Manual Valves

Several safety related valves are located in the drywell. Those in the Residual Heat Removal System, Reactor Recirculation System, Reactor Building Closed Cooling Water System, High Pressure Coolant Injection System, Reactor Core Isolation Cooling System and Core Spray System provide reactor coolant pressure boundary and/or primary containment isolation functions and do not contain non-metallic components. Valves in the CRD system performing primary containment isolation function contain non-metallic components in the seating surfaces. These valves are in the BFN local leak rate testing program and failure of the non-metallic seating surface would be detected. Total failure of the seating surface would allow some minor leakage but would not prevent the performance of the safety function.

Pneumatic Valves

The drywell has two applications of safety related pneumatic valves in Category 3 above: the reactor head vent valves and the Sampling System isolation valve. The reactor head vent valves are normally closed and fail closed. The Sampling System isolation valve is normally closed and fail closed. Non-metallic components are required to open these valves but cannot impede the closure due to failure. In both cases, the result of a failure in the non-metallic components in the operators (e.g., diaphragm, seals, etc.) would not affect the safe shutdown of the plant since the valves would fail close to perform the safety function.

Torus to Drywell Vacuum Breakers

The torus to drywell vacuum breakers are designed to control pressures between the torus and the drywell. These devices are essentially check valves. These valves contain no non-metallic components (i.e., category 1 above).

The internal process and external environmental radiation changes associated with EPU will have no adverse effect on mechanical equipment's capability to perform safety related functions.

Standby Gas Treatment System (SGTS)

The only area outside of primary containment which does not fall into one of these categories is the SGTS room. The SGTS is not subjected to normal dose. The SGTS room experiences a gamma dose due to post accident iodine loading of the charcoal filter banks. The increase in iodine loading on the SGTS filters is due to a combination of the effects of extended power uprate and the transition in fission product distribution (ORIGEN fission product distribution versus the TID 14844 fission product distribution).

A review of the safety related mechanical equipment in the SGTS room determined that the only components which could contain non-metallic parts are dampers, fan drive belts, ductwork connection boots, and the charcoal filter beds. The dampers are primarily metallic; however, there are non-metallic sealing gaskets that are used for mechanical joint integrity. Although these gaskets will deteriorate over time, they are not likely to disintegrate immediately or entirely since they are held in place and under pressure from both sides. A failure of the gaskets would allow some minor leakage but will not negate the ability of the SGTS to perform its intended safety functions.

The SGTS fans are primarily metallic with the only non-metallic part being the fan belts that drive the fan assemblies. Each fan is equipped with three parallel belts of the same size driving the same pulley. BFN maintenance procedures require periodic inspection and replacement of the drive belts. The installed belts are commercial grade items which when subjected to the radiation levels in the area may lose their strength but are unlikely to all fail catastrophically at the same time. Therefore, the belts will perform their intended functions.

The ductwork connection boots are designed to provide a flexible coupling from one section of ductwork to another. The installed boots are made from a neoprene coated nylon fabric which has been determined to be resistant to radiation dose increases with exposure to high radiation; they experience elongation and gradual degradation of overall properties. With no failure of integrity, the connection boots are expected to perform their safety functions.

The charcoal is a nonmetallic material. The purpose of the charcoal is to collect iodine. An increase in dose will not affect the ability of the charcoal bed to perform its function.

Effects of Humidity

No change will occur as a result of operating at extended power uprated conditions to the moisture concentration which is presently utilized as the design basis conditions; therefore, there is no impact on the performance of safety related mechanical equipment.

Effects of Pressure

No increases in environmental pressures have been postulated for operation at extended power uprated conditions. The peak drywell pressure will remain within the design pressure of the drywell shell and within the external design pressure of equipment located within the drywell. The environmental pressure within the reactor building is controlled by the secondary containment blowout panels which are not affected by extended power uprate.

Therefore, there is no impact on the performance of safety related mechanical equipment as a result of pressures associated with extended power uprate.

Effects of Submergence

There are no significant changes (less than 1 inch) in internal flood levels resulting from extended power uprate that would affect mechanical equipment. The flood levels are controlled by the overall physical design of the plant (i.e., relative location of stairwells and hatches, height of door sills/door louvers, etc.) which are unaffected by extended power uprate. Additionally, the water sources (e.g., tank volumes) are unaffected by extended power uprate.

Conclusion

The changes to plant conditions, both process and environmental, created by extended power uprate have been assessed and determined not to prevent mechanical equipment from performing its safety related functions. Mechanical equipment has been reviewed for changes in temperature, humidity, radiation, pressure, and flooding. Mechanical equipment has been assessed for normal, abnormal and accident conditions to ensure that it will perform its safety related functions following the implementation of extended power uprate. Short term temperature excursions have been assessed and determined to pose no detrimental effect on the operation of mechanical equipment. This is based on the relatively short duration over which they occur and the time response of the material heat-up relative to the length of time for which the increase is in effect. The non-metallic items are internal to the devices and, thus shielded from the effects of short term exposures or their failure will not affect the safety related function of the equipment. Additionally, the insulation covering many of those components provides thermal shielding for the component. No change will occur in the postulated post event humidity condition. There are no significant changes in internal submergence levels. Most areas of the plant experience either an insignificant or no increase in total integrated radiation dose. For the areas found to experience a larger increase in dose, an assessment of mechanical equipment determined the equipment's ability to perform safety related functions would not be negated by the environmental effects of extended power uprate.

There are no differences between Units 1, 2 and 3 in the types of mechanical equipment or the environmental conditions that would affect the above discussions relative to mechanical equipment qualification.

NRC Request 5.c.

Enclosure 7 indicates that further evaluations may identify the need for additional modifications or obviate the need for modifications that are currently planned for implementing the proposed EPU. All evaluations in support of the proposed EPU must be completed and any modifications that are necessary for implementing the proposed EPU must be identified and evaluated pursuant to 10 CFR 50.59 requirements such that modifications that require NRC review and approval are properly identified, specifically recognized, and evaluated, if necessary, in the amendment request.

TVA Reply 5.c.

All of the evaluations needed to support Units 2 and 3 EPU operation, with the exception of the steam dryers are complete, and all modifications identified. No further EPU related license amendments are required.

NRC Request 5.d.

Enclosure 4, Section 4.2.5, should be expanded to address protective coatings. The following information was found to be missing or incomplete:

NRC Request 5.d.(1)

Discuss the effect of EPU on qualified coatings and analyses including failures of delamination of qualified and unqualified coatings (pressure, temperature, integrated dose).

TVA Reply 5.d.(1)

The BFN Service Level 1 coatings (References 3 and 4) are subject to the requirements of Regulatory Guide 1.54 - 1973, American National Standard Institute (ANSI) N101.2 - 1972 and ANSI N101.4 - 1972. The qualification testing for Service Level 1 coatings used for new applications or repair/replacement activities inside containment meets the applicable requirements contained in the standards and regulatory commitments listed above.

Previous testing was performed which bound the peal accident conditions for all but one specific coating configuration. Therefore, TVA is performing confirmatory testing to ensure that all qualified coating configurations have been tested. The gamma dose in the testing was 1×10^9 rads, which is greater than the 1.5×10^8 rads accumulated dose for the design basis accident at 120% power. The containment pressure/temperature profiles (peak values ≥70 psig/340° F) for the coating qualification exceeds the bounding calculated post accident pressure/temperature profiles (peak value 48.5 psig/295.2° F (LOCA) 336° F (main steam line break)).

The environmental conditions under EPU are bounded by previous and confirmatory testing environmental qualification testing and therefore there is no effect on the performance of qualified coatings at EPU conditions.

All unqualified coatings are assumed to dis-bond and be available for Emergency Core Cooling System (ECCS) suction strainer blockage; therefore, changing the environmental conditions has no impact on the performance and/or failure of unqualified coatings. The quantity of unqualified coating in the drywell is tracked to ensure the total amount is less than that assumed in the ECCS suction strainer calculations.

NRC Request 5.d.(2)

Discuss whether original qualification standards for Service Level 1 coatings are still bounding under EPU conditions.

TVA Reply 5.d.(2)

The original qualification standards for Service Level 1 coatings continue to be bounding under EPU conditions. See TVA Reply 5.d(1) above.

NRC Request 5.d.(3)

Discuss the effect of EPU on "zone of influence" during a postulated design-basis accident. Discuss whether EPU will result in an increase in the failure of qualified coatings.

TVA Reply 5.d.(3)

The reactor coolant system operating temperature and pressure for EPU are the same as the conditions at the current thermal power. The BFN acceptance criteria for protective coating systems are based on NEDO-32686, "BWROG Utility Resolution Guidance (URG) for ECCS Suction Strainer Blockage." In this report analysis, the jet impingement of qualified containment coatings has been bounded by assuming a 24 inch diameter pipe break removes 100 percent of the containment coating from the drywell wall at a distance of 20 feet (10 pipe diameters) from the break. Even though the generation of transportable debris due to jets may extend beyond 10 pipe diameters, the debris generation zone is limited to 20 feet because the jet itself would be fully intercepted by major drywell structures (piping, pipe supports, grating, etc) and thus dispersed. Furthermore, the analysis assumes that since there could be pipe hangers, structural steel, valves, or other coated items in the jet path within the jet cone, the surface area of affected coating is doubled in the analysis. The 1035 psig pressure and associated 551° F temperature has no effect on the amount of coatings stripped by the jet due to the 20 foot jet modeling methodology in the original URG determination of the zones of influence. The break spectrum which includes breaks from double ended guillotine suction and discharge of the largest pipe down to very small breaks is unchanged from the current thermal power to 120% power. Therefore, EPU effects on jet impingement and the zone of influence is bounded by the degree of conservatisms in the original determination of drywell debris.

NRC Request 5.e.

Enclosure 4, Section 6.4, Water Systems, does not address nonsafety-related loads in the service water system.

TVA Reply 5.e.

Enclosure 4 of the license amendment application, Section 6.4, Water Systems, addresses both safety-related and nonsafety-related loads in the service water system.

Safety-related service water systems are addressed in Sections 6.4.1.1 (Safety-Related Loads), 6.4.1.1.1 (Emergency Equipment Cooling Water System), 6.4.1.1.2 (Residual Heat Removal Service Water System), and 6.4.5 (Ultimate Heat Sink).

Nonsafety-related service water systems are addressed in Sections 6.4.3 (Reactor Building Closed Cooling Water System) and 6.4.4 (Raw Cooling Water system).

These Enclosure 4 sections provide the information for each of the systems regarding acceptable performance with the plant operating at EPU conditions.

NRC Request 5.f.

Enclosure 4, Section 6.1, Electrical Power and Auxiliary Systems, Section 9.3.2, Station Blackout, and Section 10.3.1, Environmental Qualification for Electrical Equipment should be expanded to address the physical modifications that will need to be made to address the uprated capacity as well as unique and multi-unit features. Additionally, a discussion on the effects for Unit 2 should be included.

TVA Reply 5.f.

The EPU related modifications required to address the uprated capacity to the off-site power distribution system are discussed in Enclosure 4 to the license amendment request Section 6.1.1. The following modifications will be performed:

5) The main isolated phase bus duct for each unit is being uprated to have a continuous current rating and asymmetrical current rating sufficient to support the generator at EPU conditions. This uprate necessitates an upgrade to the bus duct cooling system,

6) The tap isolated phase bus duct is being uprated to have a asymmetrical current rating sufficient to support the generator at EPU conditions. TVA has determined that no physical modifications are required,

7) The generator breaker for each unit is being modified to have a continuous current rating and asymmetrical current rating sufficient to support the generator at EPU conditions,

8) The three main transformers for each unit will be replaced along with the installation of two spares for the site.

There are no multi-unit impacts on the off-site or on-site power distribution systems due to simultaneous operation or accident mitigation of Units 1, 2 and 3 for operation at the uprated conditions.

Enclosure 4 to the license amendment request, Section 9.3.2 discusses station blackout (SBO) and states there are no changes to the systems and equipment used to respond to an SBO, nor is the required coping time changed. Therefore, no modifications are required for SBO.

Enclosure 4 to the license amendment request, Section 10.3.1.1 discusses the environmental qualification for electrical equipment inside containment and Section 10.3.1.2 discusses the environmental qualification for electrical equipment outside containment. As discussed in Enclosure 4, Sections 10.3.1.1 and 10.3.1.2, the environmental impact of EPU operation is minimal compared to conditions at 105% of OLTP. Therefore, there are no modifications for electrical equipment qualification due to the change to EPU operation.

<u>NRC Request 5.q.</u>

Enclosure 4, Section 3.4, should be expanded to address the potential for recirculation pump seizure and/or a recirculation pump shaft break.

<u>TVA Reply 5.q.</u>

NRC has previously accepted the list of transients to be evaluated in Table E-1 of ELTR1 and the Recirculation Pump Rotor Seizure and The Recirculation Pump Shaft Break (events that result in a decrease in reactor core coolant flow rate) are not in the events required by the ELTR.

It is important to note that Recirculation Pump Rotor Seizure and Recirculation Pump Shaft Break are events that result in a decrease in reactor core coolant flow rate. For these two events, the Recirculation Pump Rotor Seizure is more severe because a single Recirculation Pump is assumed to stop instantaneously resulting in a quicker reduction in core coolant flow rate than for a Recirculation Pump Shaft Break.

Although not required by the ELTR, fuel specific analyses at EPU operating conditions have been performed for the Recirculation Pump Seizure event. This is a very mild event in relation to other accidents such as LOCA since following the event, although one recirculation driving loop flow is lost, there is no inventory loss, core flow continues, water level is maintained, and the core remains submerged providing a continuous core cooling mechanism.

EPU fuel specific analysis reflects that peak neutron and heat fluxes do not increase above initial conditions and reactor pressure does not change significantly. GE and FANP analysis demonstrate that the results for the pump seizure event are significantly less severe than the consequences of a LOCA event. The resulting increased temperature of the cladding and reduced reactor pressure for a LOCA combine to yield a much more severe stress and potential for cladding perforation than for the recirculation pump seizure event. Therefore, this event is not limiting and the potential effects are very conservatively bounded by the effects of a LOCA.

NRC Request 5.h.

Enclosure 4 should be expanded to address uncontrolled control rod assembly withdrawal from a subcritical or low power startup condition.

TVA Reply 5.h.

Continuous Rod Withdrawal during a reactor startup from a subcritical or low power startup condition is described in Browns Ferry UFSAR Section 14.5.4.2. As described in the UFSAR, the most severe consequence for this event would occur when the reactor is just critical at room temperature and an out-of-sequence rod is continuously withdrawn.

At 100% OLTP conditions, calculations determined that a fuel enthalpy of 60 cal/gm would result. Peak fuel enthalpy is not changed significantly by EPU for this low power cold operating condition and therefore this event did not need to be reanalyzed for extended power uprate due to the amount of margin to the fuel failure threshold of 170 cal/gm. Furthermore, since this is a cold event and the cold and low power reactivity characteristics of the fuel are unaffected by extending the allowable maximum operating power of the core, the consequences of this event are not affected by power uprate.

The NRC Staff Position for ELTR1 states in Appendix E, Section 2.4 that only the limiting transients need be included in the uprate amendment request, but a list of all transients analyzed in support of power uprate should be included. As stated in ELTR1 Section E.2.2, the minimum list of events to be included in the power uprate evaluation (Table E-1) confirms that the existing set of reload analysis transients remain valid for power uprate. In regards to the limiting control rod withdrawal event, the Rod Withdrawal Error (RWE) event identified in the ELTR table is the power operation RWE (as opposed to the low power RWE) since it is the only RWE that can potentially challenge a fuel design parameter. The power operation RWE has been analyzed and the results provided in Enclosures 4 and 5, Table 9-2 of the PUSAR and FUSAR.

NRC Request 5.i.

Enclosure 4 should be expanded to address the inadvertent opening of a boiling-water reactor pressure relief valve.

TVA Reply 5.i.

As indicated in ELTR1, bounding transients are documented in GESTAR and Table E-1 of ELTR1 includes those events plus a few additional cases intended to reconfirm the limiting events on a plant-specific basis. The transient events in Table E-1 were analyzed for the BFN Extended Power Uprate and results are reported in Table 9-2 of Enclosure 4 and 5 of the license amendment request. The inadvertent opening of a BFN pressure relief valve is described in the BFN UFSAR Section 14.5.5.2. This is a mild depressurization transient which is not a limiting transient event and is thus, not required to be reanalyzed. However, this event has been analyzed at Extended Power Uprate conditions. At EPU conditions, the inadvertent opening of one of the relief valves continues to produce a mild depressurization transient. The turbine pressure regulator senses the pressure decrease and reduces turbine flow and the reactor settles close to the pre-event power. GE and FANP analyses demonstrate that the peak neutron flux and fuel surface heat flux do not exceed the initial power values and no fuel damage results. This event is bounded by the loss of feedwater event whose EPU analysis results are reported in Enclosure 4, Table 9-2.

References:

1. TVA letter to NRC dated November 21, 2002: "Browns Ferry Nuclear Plant (BFN) – Units 2 and 3 – Options for Extended Power Uprate (EPU) and Fuel Vendor Change."

2. NRC letter to TVA dated January 17, 2003: "Browns Ferry Nuclear plant, Units 2 and 3 – Fuel Vendor Change, Extended Power Uprate, and Maximum Extended Load Line Limit Analysis Plus Submittals."

3. TVA letter to NRC dated November 10, 1998: "Browns Ferry Nuclear Plant (BFN), Sequoyah Nuclear Plant (SQN), and Watts Bar Nuclear Plant (WBN), 120 day response to Generic letter (GL) 98-04 Potential for Degradation of Emergency Core Cooling System (ECCS) after a Loss of Coolant (LOCA) Because of Construction and Protective Coating Deficiencies and Foreign material in Containment."

4. TVA letter to NRC dated May 11, 2004: "Browns Ferry Nuclear Plant (BFN) Unit 1 response to NRC Generic Letter (GL) 98-04, Potential for Degradation of the Emergency Core Cooling System (ECCS) after a Loss-of-Coolant (LOCA) Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment."

5. TVA letter to NRC dated January 24, 2005: "Tennessee Valley Authority – Browns Ferry Nuclear Plant (BFN) – Unit 3 – Docket 50-296 – Facility Operating License DPR – 68 – License Event Report (LER) 50-296/2004-002-00."