TVA-BFN-431

March 7, 2006

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 No. 50-259
Tennessee Valley Authority	)	

BROWNS FERRY NUCLEAR PLANT (BFN) - UNIT 1 - RESPONSE TO NRC ROUND 3 REQUESTS FOR ADDITIONAL INFORMATION RELATED TO TECHNICAL SPECIFICATIONS (TS) CHANGE NO. TS-431 - REQUEST FOR EXTENDED POWER UPRATE OPERATION (TAC NO. MC3812)

This letter provides TVA's response to the NRC Staff's request for additional information, which was submitted to TVA by letter dated December 22, 2005 (ADAMS Accession No. ML053560120), in order to support review of the BFN Unit 1 Extended Power Uprate (EPU) license amendment application.

TVA submitted the BFN Unit 1 EPU application to the NRC by letter dated June 28, 2004 (ML041840109). TVA supplemented that application by letters dated August 23, 2004 (ML042370849), February 23, 2005 (ML050560150), April 25, 2005 (ML051170244), June 6, 2005 (ML051580249), and February 28, 2006. Enclosure 1 to this letter provides TVA's responses to the NRC requests.

Enclosure 2 to this letter contains revised responses to four of the requests answered in TVA letter dated December 19, 2005 (ML053560194).

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Some of the information in Enclosure 1 is proprietary to General Electric Nuclear Energy (GENE). GENE 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 in Enclosure 1. A non-proprietary version of this response is contained in Enclosure 8. Additionally, Enclosures 9 and 10 provide proprietary and non-proprietary versions, respectively, of licensing topical report NEDC-33173P.

During preparation and final review of this submittal, a legacy error was discovered in the existing design calculation which determines the available Emergency Core Cooling System pump net positive suction head requirements. The error has been documented in BFN's Corrective Action Program, and the calculation is presently being revised. The effect of the error is small; however, it impacts numerical values that were provided in the original EPU submittal and in the February 28, 2006 submittal. Additionally, the error impacts values that are needed to respond to questions ACVB.17, ACVB.18, ACVB.26, and ACVB.32. Therefore, the responses to these questions with the corrected information will be provided in a separate letter by March 24, 2006. This issue was discussed with Margaret Chernoff on March 6, 2006.

There are no new regulatory commitments associated with this submittal. If you have any questions concerning this letter, please contact me at (256) 729-2636.

I declare under penalty of perjury that the foregoing is true and correct. Executed on this 7th day of March, 2006.

Sincerely,

William D. Crouch
Manager of Licensing
and Industry Affairs

cc: See page 5.

Enclosures:

1. Response To December 22, 2005, NRC Round 3 Requests For Additional Information Related To Technical Specifications (TS) Change No. TS-431 - Request For Extended Power Uprate Operation (Proprietary Version)

2. Revised Responses To TVA Submittal Dated December 19, 2005, Related To Technical Specifications (TS) Change No. TS-431 - Request For Extended Power Uprate Operation

3. EPU Power Ascension Test Plan

4. May 23, 1975 - Final Summary Report, Unit 2 Startup, Browns Ferry Nuclear Plant

5. May 9, 1977 - Final Summary Report, Unit 3 Startup, Browns Ferry Nuclear Plant

6. RS-001 Revised Template Safety Evaluation

7. Copies of Material Provided To U.S. Fish and Wildlife Service

8. Response To December 22, 2005, NRC Round 3 Requests For Additional Information Related To Technical Specifications (TS) Change No. TS-431 - Request For Extended Power Uprate Operation (Non-Proprietary Version)

9. NEDC-33173P, February 2006, "Applicability of GE Methods to Expanded Operating Domains" (Proprietary Version)

10. NEDC-33173, February 2006, "Applicability of GE Methods to Expanded Operating Domains" (Non-Proprietary Version)

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ENCLOSURE 2

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNIT 1

REVISED RESPONSES TO TVA SUBMITTAL DATED DECEMBER 19, 2005 RELATED TO TECHNICAL SPECIFICATIONS (TS) CHANGE NO. TS-431 - REQUEST FOR EXTENDED POWER UPRATE OPERATION

NRC Request EMCB-A.1

Section 10.7, Plant Life, in Enclosure 4 of the June 25, 2004, submittal, identifies irradiation-assisted stress-corrosion cracking (IASCC) as a degradation mechanism influenced by increases in neutron fluence and reactor coolant flow. This section indicates that the current inspection strategy for reactor internal components is expected to be adequate to manage any potential effects of EPU operating conditions. Note 1 in Matrix 1 of Section 2.1 of RS-001, Revision 0 indicates that guidance on the neutron irradiation-related threshold for IASCC in boiling-water reactors (BWRs) is in Boiling-Water Reactor Vessel and Internals Program (BWRVIP) report BWRVIP-26. The "Final License Renewal SER [Safety Evaluation Report] for BWRVIP-26," dated December 7, 2000, states that the threshold fluence level for IASCC is 5×10^{20} n/cm2 (E > 1 MeV).

Identify the vessel internal components whose fluence, at the end of period of operation with the EPU operating conditions will exceed the threshold level and become susceptible to cracking due to IASCC. For each vessel internals component that exceeds the IASCC threshold, either provide an analysis that demonstrates failure of the component will not result in the loss of the intended function of the reactor internals or identify the inspection program to be utilized to manage IASCC of the component. Identify the scope, sample size, inspection method, frequency of examination and acceptance criteria for the inspection programs.

After review of the response to this request, the NRC informally noted "The staff has determined that a more detailed response to the original question is required regarding the top guide and core plate holddown bolts. Because these two components exceed the threshold of 5×10^{20} n/cm2, TVA is requested to identify the scope, sample size, inspection method, frequency of examination and acceptance criteria for the inspection programs of the top guide and core plate holddown bolts for BF, Units 1, 2, and 3.

The staff requests that TVA provide these additional details as they are not provided in the BWRVIP documents."

TVA Reply to EMCB-A.1

The requested information is provided in Enclosure 1 of this letter by the reply to EMCB-A.4. Additional information regarding the core plate holddown bolts is provided in Enclosure 1 of this letter by the reply to EMCB-A.3.

This response supplements the original response.

NRC Request SPLB-B.1

Discuss whether any administrative controls or fire protection responsibilities of plant personnel are affected by an increase in decay heat. Also, address why an increase in decay heat will not result in an increase in the potential for a radiological release from a fire.

After review of the response to this request, the staff informally noted "Still needs to address why the EPU does not affect the elements of their fire protection program related to the fire protection responsibilities of plant personnel."

TVA Reply to SPLB-B.1

Administrative controls and fire protection responsibilities of plant personnel in the Technical Specifications, the Technical Requirements Manual, the Nuclear Quality Assurance Plan, and the Fire Protection Report were reviewed for effects associated with the increase in decay heat. There are no administrative controls or fire protection responsibilities of plant personnel affected by an increase in decay heat associated with EPU.

As indicated by the results of the Appendix R analyses, all Appendix R acceptance criteria are met under EPU; therefore, there is no increase in the potential for a radiological release resulting from a fire.

This response replaces the original response.

NRC Request SPLB-B.2

Section 6.7.1, of Enclosure 4 of the June 28, 2004, submittal states that:

> ...a plant-specific evaluation was performed to demonstrate safe shutdown capability in compliance with the requirements of 10 CFR 50 Appendix R assuming EPU conditions.... The results of the Appendix R evaluation for EPU provided in Table 6-5 demonstrate that fuel cladding integrity, reactor vessel integrity, and containment integrity are maintained and that sufficient time is available for the operator to perform the necessary actions.

Upon reviewing Table 6-5, Browns Ferry Appendix R Fire Event Evaluation Results, the NRC staff was able to find references for all but the following values in the EPU submittal:

- Cladding Heatup (peak clad temperature (PCT)), degrees F = 1428 (EPU)
- Suppression Pool Bulk Temperature, degrees F = 227 (EPU), ≤ 227 (Appendix R Criteria), including Note 3
- Primary Containment Pressure, pounds per square inch gage = 13.6 (EPU)

Provide references, including appropriate extracts from the UFSAR, plant-specific Appendix R evaluation, etc., for these values in Table 6-5, including Note 3.

After review of the response to this request, the staff informally requested "If the referenced, but not provided, extracts from BFN Calculation MDN-0999-980113, App. R FP Evaluation, indicate the same PUSAR Table 6-5 values cited in the RAI for, then TVA should provide the extracts as copies or as quotes."

TVA Reply to SPLB-B.2

BFN calculation MDN0999980113, "Appendix R Fire Protection Evaluation," documents the EPU evaluation on compliance with the requirements of 10 CFR 50 Appendix R which was performed in Project Task Report T0611, "Appendix R Fire Protection." The limiting EPU PCT occurs for Case 1 and is presented in Project Task Report T0611 Section 3.3.1, "Key Results," Item 1 as 1428°F. The EPU suppression pool bulk temperature is the same for Cases 1, 2, and 3 and is presented in Project Task Report T0611 Section 3.3.1, "Key Results," Items 4, 9, and 14 as 227°F. The torus attached piping limit for EPU is the suppression pool

bulk temperature of 227°F and is the temperature used in the analyses for the torus attached piping for EPU Appendix R conditions. The primary containment pressure is the same for Cases 1, 2, and 3 and is presented in Project Task Report T0611 Section 3.3.1, "Key Results," Items 5, 10, and 15 as 13.6 psig.

This response replaces the original response.

NRC Request SPSB-A.18

Address the questions in the SRP, Chapter 19, Table III-1 concerning low power and shutdown PRA.

Following an EPU PRA Audit in January of 2006, the NRC informally noted that TVA did not answer this question as requested.

TVA Reply to SPSB-A.18

Shutdown safety is maintained and monitored by compliance with work in accordance with the outage schedule/plan. An assessment is performed of the outage schedule/plan implementation prior to the outage and, during the execution of the schedule/plan, anytime the outage schedule/plan is affected. These assessments are performed using the EPRI sponsored program called Outage Risk Assessment Management (ORAM). ORAM is a computer program that receives data from the scheduling software and performs deterministic risk assessments during reactor shutdowns and outages. The implementation of the program is controlled by procedures and includes the ORAM software that takes the status (i.e., available, unavailable) of key plant equipment, evaluates the current/planned plant condition(s) against approved data models, and then produces an output of the relative level of safety/defense in depth of key shutdown functions:

- Decay Heat Removal,
- Inventory Control,
- Electrical Power Availability,
- Reactivity Control,
- Fuel Pool Cooling, and
- Primary/Secondary Containment.

The program includes a structured approach to determine the effect of outage activities upon the key shutdown safety functions by assessing the following:

- Identify key safety functions affected by the Structure, System, and Component (SSC) planned for removal from service
- Consider the degree to which removing the SSC from service will affect the key safety functions
- Consider degree of redundancy, duration of out-of-service condition, and appropriate compensatory measures, contingencies, or protective actions that could be taken if appropriate for the activity under consideration.

An integral part of an outage schedule/plan is the contingency plan. This is an approved plan for compensatory actions:

- To maintain Defense in Depth by alternate means when outage planning reveals that specific SSCs will not be available
- To restore Defense in Depth when systems availability drops below previously established levels during the outage
- To minimize the likelihood of the loss of key safety functions during higher risk evolutions

The Shutdown Risk Assessment Program also includes a detailed review of the outage schedule/plan (including review of changes) by a multi-discipline team with extensive experience in the operation and maintenance activities at BFN. This activity provides another level of assurance that shutdown safety issues are addressed and all reasonable actions have been taken to minimize shutdown risk.

The review considers, for example:

- Technical Specifications Requirements
- The degree of redundancy available for performance of the key safety functions served by out-of-service SSCs
- The duration of the activity
- The likelihood of an initiating event or accident that would require the performance of the affected safety function
- The likelihood that the activity will increase the frequency of an initiating event requiring key safety functions
- Component and system dependencies that are affected
- Performance issues for the in service redundant SSCs
- The risk impact of performing the maintenance during shutdown with respect to performing the maintenance at power

Another important feature of the BFN shutdown risk program is the inherent flexibility that is provided by the structure of the program. Calculations are prepared that provide BFN specific information into the program. The calculations address any chances produced by the operating history of BFN prior to the outage, including power levels and durations. These calculations provide input into functional parameters such as the availability of systems and support systems required to provide reactor vessel makeup water consistent with the decay heat generation load and availability of alternate sources of reactor vessel makeup water consistent with the decay heat generation rate. This work also provides input regarding times associated with the reactor vessel and fuel pool boil down rates. This information also provides insights for determining operator response times as an integral part of this pre-outage work.

Shutdown events include the following major categories:

- Abnormal Operating Transient
  - Shutdown cooling malfunction
  - Inadvertent pump start *
  - Control rod withdrawal error *
  - Fuel assembly insertion *
  - Control rod removal *
  - Inadvertent opening of a relief valve *
  - Total loss of off-site power
  - Startup of idle recirculation pump *
  - Loss of shutdown cooling

- Accident
  - Fuel-handling *

- Special Event
  - Loss of habitability of the control room *
  - Ability to shutdown reactor without control rods *

* These postulated event impacts and associated mitigating SSCs (including operator actions) are not affected by the implementation of Extended Power Uprate (EPU). BFN operation at a higher power level will not affect cool water effects, SSCs performance regarding operational capability, environmental heat load, or interfere with operator actions designed to assist with the mitigation of these postulated events.

For the remaining three events, BFN operation at EPU conditions will have a very minor affect regarding mitigation of the postulated events during shutdown conditions.

- Shutdown cooling malfunction
- Total loss of off-site power
- Loss of shutdown cooling

For these events, EPU operation does not affect equipment reliability, availability, initiating event frequency, and mitigation approach including equipment utilized for mitigation. The effect of EPU operation on the success criteria is similar to the effect on the at power PRA success criteria. However, because the reactor has been shutdown for some period of time, the decay heat load is substantially lower than the at power values. This situation results in boil down times that are much longer than the values associated with the at power conditions. These conditions result in small or no changes in the success criteria of systems associated with mitigation of events postulated during shutdown conditions at BFN. There is an effect on mitigation associated with decay heat load and the resulting effect on operation actions. The BFN use of ORAM appropriately addresses this aspect regarding event mitigation by including calculations that are cycle specific and address previous operating power levels and associated durations. These calculations provide input into functional parameters such as the availability of systems and support systems required to provide reactor vessel makeup water consistent with the decay heat generation load and availability of alternate sources of reactor vessel makeup water consistent with the decay heat generation rate. This information reflects operator action response times also. The reduction for these operator action times due to EPU operation is shown to be less than 15% (depending on the time after shutdown). These small changes in already relatively lengthy operator response times result in negligible changes in human action probabilistic values.

BFN plans to continue the use of ORAM as a tool to provide for a continued structured program associated with the outages schedule/plan.

Using this information, the following SRP questions are answered.

- Does the application introduce new initiating events or change the frequencies of existing events?

  No new initiating events or increased potential for initiating events during shutdown are postulated due to the proposed EPU.

- Does the application affect the scheduling of outage activities?

  No. BFN operation at EPU conditions will not change the outage sequence of operations to accomplish shutdown activities. Decay heat loads will increase due to EPU conditions but this minimal effect will be anticipated and appropriately planned for by the ORAM pre-outage schedule/plan.

- Does the application affect the ability of the operator to respond to shutdown events?

  No. The BFN use of ORAM appropriately addresses this aspect regarding event mitigation by including calculations that are cycle specific and address previous operating power levels and associated durations. These calculations provide input into functional parameters such as the availability of systems and support systems required to provide reactor vessel makeup water consistent with the decay heat generation load and availability of alternate sources of reactor vessel makeup water consistent with the decay heat generation rate. This information reflects operator action response times also. The reduction for these operator action times due to EPU operation is shown to be less than 15% (depending on the time after shutdown). These small changes in already relatively lengthy operator response times result in negligible changes in human action probabilistic values.

- Does the application affect the reliability or availability of equipment used for shutdown conditions?

  No. For these events, EPU operation does not affect equipment reliability, availability, initiating event frequency, and mitigation approach including equipment utilized for mitigation.

- Does the application affect the availability of equipment or instrumentation used for contingency plans?

  No. Consistent with the situation associated with the unaffected reliability or availability of equipment used for shutdown conditions, equipment and instrumentation reliability, availability, initiating event frequency, and mitigation approach associated with the contingency plan will not be effected by EPU operation.

This response replaces the original response.

ENCLOSURE 3

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNIT 1

EPU POWER ASCENSION TEST PLAN

Table 1 contains a listing of the currently planned modifications necessary to support EPU that require testing during power ascension. A description of each activity and the planned testing is provided. Required post modification testing that will be performed prior to power ascension in accordance with the plant design change process is also provided in Table 1. Modifications that are required for EPU that are not tested during power ascension are not listed. Setpoint adjustments, including those required for Unit 1 due to the steam dome pressure increase, that are tested by standard plant procedures such as required Technical Specifications surveillance tests are not listed.

Table 2 contains a list of planned power ascension tests that are required to specifically address EPU implementation. EPU testing performed by standard plant procedures as a part of normal startup testing are not listed.

Table 3 describes the BFN EPU Power Ascension Test Plan.

The modifications and testing activities in Tables 1, 2, and 3 represent the currently planned post modification tests and power ascension test activities. Details of some testing activities may be modified based on further evaluation.

Table 1 - BFN EPU Planned Modifications Requiring Power Ascension Testing		
Activity	**Description**	**Testing**
Main Turbine	• Replace HP Turbine diaphragms and rotor buckets • Replace HP Rotor/LP Rotors (Unit 1 only) • Replace springs, bonnets, washers, bellows, & bolting on six 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	• Monitor steam seal header pressure • Calibration of the steam seal header pressure controller • Inservice leak test

Table 1 - BFN EPU Planned Modifications Requiring Power Ascension Testing		
Activity	**Description**	**Testing**
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 Pump trip test
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 • Condensate Booster Pump trip test

Table 1 - BFN EPU Planned Modifications Requiring Power Ascension Testing		
Activity	**Description**	**Testing**
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 Pump trip test
Moisture Separators	• Change vanes and add perforated plate on moisture separators • Modify internal drains as needed	• Moisture removal effectiveness testing • Inservice leak test • Performance monitoring (flow, pressure, temperatures, etc.)

Table 1 - BFN EPU Planned Modifications Requiring Power Ascension Testing		
Activity	**Description**	**Testing**
Feedwater Heaters	• Upgrade heater shell pressure certification • Rerate tube side pressure certification for feedwater heater (FWH) 3 • 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 / re-weld pass partition plates in all FWHs • Install manway stiffeners on FWH 3	• Relief valve bench testing • Performance monitoring (flow, pressure, temperatures, etc.) • Instrumentation calibration and functional testing • Inservice leak test
Condensate Demineralizers	• Install 1 new vessel with valves & digital controls • Upgrade controls on 9 existing vessels to digital (Unit 2 only) • Install digital control on 9 existing vessels (Unit 1 only) • Replace valves for increased reliability	• Control system functional testing • Initial installation startup test (flow, pressure, temperatures, etc.)

Table 1 - BFN EPU Planned Modifications Requiring Power Ascension Testing		
Activity	**Description**	**Testing**
Steam Dryer	• Modify dryer to ensure structural integrity at EPU conditions	• Determine moisture carryover • Monitor main steam line pressure data
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

Table 1 - BFN EPU Planned Modifications Requiring Power Ascension Testing		
Activity	**Description**	**Testing**
Main Generator System	• Recalibrate/replace pressure regulators and pressure switches • Increase generator hydrogen to 75 psig to operate at increased loads • Rewind generator stator and generator field (Unit 1 only)	• Field installation testing • Instrumentation calibration and functional testing • Monitoring of system parameters (voltage, amps, temperatures, etc.) during power ascension
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
Main Bank Transformers	• Replace due to obsolescence issues. The Unit 1, Unit 2, and Unit 1/2 spare transformers are in place and operating at this time. The Unit 3 transformers are currently scheduled to be replaced in 2010 along with the installation of a dedicated spare Unit 3 transformer.	• Performance monitoring
Vibration Monitoring	• Install temporary sensors based on ongoing analyses • Conduct testing program during power ascension	• Collect and analyze vibration data on selected systems

Table 1 - BFN EPU Planned Modifications Requiring Power Ascension Testing		
Activity	**Description**	**Testing**
Main Steam Isolation Valves	• Replace MSIV poppets and modify operators (Unit 1 only) 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	• Performance monitoring
EHC Software	• New program inputs & logic for EPU conditions	• Verification of control functions • Turbine Valve setup • Controls Tuning
Steam / Feedwater Normal Flow Rate Increase	• Increased flow rate to accommodate increased reactor thermal power output	• Monitor to ensure plant remains within anticipated operational limits
Recirculation Pump Flow Rate Increase	• Increased required recirculation pump flow rate required to achieve total core flow	• Verification of total core flow

Table 2 - BFN EPU Planned Power Ascension Tests		
Corresponding Original S/U Test Number	**Test**	**Test Description**
STP 1	Chemical and Radiochemical	Sampling and measurements selected power levels to determine 1) the chemical and radiochemical quality of reactor water and reactor feedwater and 2) gaseous release.
STP 2	Radiation Measurements	Gamma dose rate measurements and where appropriate, neutron dose rate measurements at specific limiting locations throughout the plant to assess the impact of the uprate on actual plant area dose rates.
STP 10	IRM Calibration	After the APRM calibration for EPU, the IRM gains will be adjusted as necessary to assure the IRM overlap with the APRMs. This will be done during first controlled shutdown following APRM calibration for EPU.
STP 17 (Unit 1 only)	System Expansion	Due to the 30 psi reactor pressure increase (and associated temperature increase), system expansion checks will be made for major equipment and piping in the nuclear steam supply system during heatup to assure components are free to move as designed and adjustments will be made as necessary for freedom of movement.

Table 2 - BFN EPU Planned Power Ascension Tests		
Corresponding Original S/U Test Number	**Test**	**Test Description**
STP 19	Core Performance	Core performance parameters (LHGR, APLHGR, and MCPR) will be calculated to verify they remain within limits as part of a careful, monitored approach to the EPU power level.
STP 20	Electrical Output and Heat Rate	Demonstrate that the plant net electrical output and net heat rate requirements are satisfied.
STP 22	Pressure Regulator	Evaluate pressure control system response to pressure setpoint testing.
STP 23	Feedwater Control System	Adjust the Feedwater Control System for acceptable reactor water level control. Demonstrate the capability to prevent a low reactor water level scram following the trip of a single condensate pump, condensate booster pump, or feedwater pump.
STP 92	Steam Separator-Dryer	Determine steam separator-dryer moisture carryover.

Table 3 - BFN EPU Power Ascension Test Plan																												
Test / Modification	Test Descrip-tion[1]	Prior to Startup	EPU TEST CONDITIONS - PERCENT OF 3293 MWT (OLTP) (Allowance +0% -5%)																				(Allowance +0% - 1%)				
			0	5	10	15	20	25	30	35	40	45	50	55	60	65	70	75	80	85	90	95	100	105	110	115	EPU
Main Turbine	Table 1	X						X															X	X	X	X	X
Turbine Sealing System	Table 1		Performance monitoring from 0% to EPU																								
Condensate Pumps	Table 1	X	Performance monitoring from 0% to EPU																								
Condensate Booster Pumps	Table 1	X	Performance monitoring from 0% to EPU																								
Feedwater Pumps and Turbines	Table 1	X	Performance monitoring from 0% to EPU																								
Moisture Separators	Table 1												X					X					X	X	X	X	X
Feedwater Heaters	Table 1	X											X					X					X	X	X	X	X
Condensate Demineralizers	Table 1	X																									
Steam Dryer	Table 1																						X	X	X	X	X
Reactor Recirc Pump Motors	Table 1	X																					X	X	X	X	X
Main Generator System	Table 1	X						Performance monitoring from 25% to EPU																			

Table 3 - BFN EPU Power Ascension Test Plan																												
Test / Modification	Test Descrip-tion[1]	Prior to Startup	EPU TEST CONDITIONS - PERCENT OF 3293 MWT (OLTP) (Allowance +0% -5%)																				(Allowance +0% - 1%)				
			0	5	10	15	20	25	30	35	40	45	50	55	60	65	70	75	80	85	90	95	100	105	110	115	EPU
Isolation Phase Bus Duct Cooling	Table 1							Performance monitoring from 25% to EPU																			
Main Bank Transformers	Table 1																						X	X	X	X	X
Vibration Monitoring	Table 1												X					X					X	X	X	X	X
Main Steam Isolation Valves	Table 1																						X	X	X	X	X
EHC Software	Table 1							X					X			X		X		X		X	X	X	X	X	X
Steam/Feedwater Normal Flow Rate	Table 1																						X	X	X	X	X
Recirculation Pump Flow Rate	Table 1																						X	X	X	X	X
Chemical and Radiochemical	Table 2																						X	X	X	X	X
Radiation Measurements	Table 2																						X	X	X	X	X
IRM Calibration	Table 2	(Not a startup test. Will be performed during the first controlled shutdown following APRM calibration for EPU.)																									

Table 3 - BFN EPU Power Ascension Test Plan																												
Test / Modification	Test Descrip-tion[1]	Prior to Startup	EPU TEST CONDITIONS - PERCENT OF 3293 MWT (OLTP) (Allowance +0% -5%)																				(Allowance +0% - 1%)				
			0	5	10	15	20	25	30	35	40	45	50	55	60	65	70	75	80	85	90	95	100	105	110	115	EPU
System Expansion (Unit 1 only)	Table 2	X	X																								
Core Performance	Table 2																						X	X	X	X	X
Electrical Output and Heat Rate	Table 2																										X
Pressure Regulator	Table 2							X					X			X		X		X		X	X	X	X	X	X
Feedwater Control System	Table 2																						X	X	X	X	X
Steam Separator-Dryer	Table 2																						X	X	X	X	X

[1] Line items may have multiple tests. Each test will not necessarily be performed at every power level indicated.

ENCLOSURE 4

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNIT 1

MAY 23, 1975 - FINAL SUMMARY REPORT, UNIT 2 STARTUP
BROWNS FERRY NUCLEAR PLANT

ENCLOSURE 5

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNIT 1

MAY 9, 1977 - FINAL SUMMARY REPORT, UNIT 3 STARTUP
BROWNS FERRY NUCLEAR PLANT

ENCLOSURE 6

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNIT 1

EXTENDED POWER UPRATE RS-001 REVISED TEMPLATE SAFETY EVALUATION

The attached pages have been revised. On the affected pages, the revised portions have been highlighted. A line has been drawn through the deleted text and a double underline for new or revised text.

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; and (3) draft GDC-6, insofar as it requires that decay heat removal systems shall be provided for all expected conditions of normal operation. 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, 6, 40 and 42 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the RHR system.

C-01

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-9, insofar as it requires that the reactor coolant pressure boundary shall be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime; (~~2~~3) 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~~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 15.1.1-4 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

C-02

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, 9, 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.

C-03

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; (2) draft GDC-9, insofar as it requires that the reactor coolant pressure boundary shall be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime; and (~~2~~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 15.2.1-5 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

E-04

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, 9, 27, and 28 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the events stated.

C-05

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; (2) draft GDC-9, insofar as it requires that the reactor coolant pressure boundary shall be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime; and (~~2~~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 15.2.6 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

C-06

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, 9, 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.

C-07

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; (2) draft GDC-9, insofar as it requires that the reactor coolant pressure boundary shall be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime; and (~~2~~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 15.2.7 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

C-8

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, 9, 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.

C-09

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; (2) draft GDC-9, insofar as it requires that the reactor coolant pressure boundary shall be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime; and (~~2~~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 15.3.1-2 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

C-10

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, 9, 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.

C-11

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-9, insofar as it requires that the reactor coolant pressure boundary shall be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime; (~~2~~3) 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~~4) 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~~5) 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, 9, 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.

C-13

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; (2) draft GDC-9, insofar as it requires that the reactor coolant pressure boundary shall be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime; and (~~2~~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 15.5.1-2 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

C-14

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, 9, 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; (2) draft GDC-9, insofar as it requires that the reactor coolant pressure boundary shall be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime; and (~~2~~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 15.6.1 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

C-16

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, 9, 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.

C-17