April 13, 2006

TVA-BFN-TS-418
TVA-BFN-TS-431

10 CFR 50.90

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
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Washington, D. C. 20555-0001

Gentlemen:

In the Matter of	)	Docket Nos. 50-259
Tennessee Valley Authority	)	50-260
	)	50-296

BROWNS FERRY NUCLEAR PLANT (BFN) – UNITS 1, 2, AND 3 - TECHNICAL SPECIFICATIONS (TS) CHANGE NOS. TS-418 AND TS-431 – EXTENDED POWER UPRATE (EPU) OPERATION - REVISED RESPONSES TO NRC ROUND 2 REQUESTS FOR ADDITIONAL INFORMATION - (TAC NOS. MC3812, MC3743, AND MC3744)

By letters dated December 19, 2005 (ADAMS Accession Nos. ML053560194 and ML053560186) TVA submitted responses to NRC Round 2 requests for additional information (RAIs) regarding TVA's applications for extended power uprate of BFN Unit 1 and BFN Units 2 and 3, respectively. As a result of discussions with the NRC staff, TVA is revising its responses to five of the RAIs. The responses to the subject RAIs are the same for all three BFN units.

Enclosure 1 to this letter provides revised responses to RAIs IPSB-B.1 and IPSB-B.8 regarding external radiation doses due to direct radiation and skyshine. Enclosure 2 to this letter provides revised responses to RAIs SPLB-A.1, SPLB-A.2 and SPLB-A.3 regarding fuel pool cooling. Each revised RAI response in Enclosure 1 supersedes the response previously provided to the NRC staff. However, the responses in Enclosure 2 supplement the previous responses to the respective RAIs.

Item 3.1.2(5) of the Supplemental Reply to RAIs SPLB-A.1, 2, and 3 in Enclosure 2 is not complete. As discussed with the NRC staff on April 13, 2006, TVA will address this item in a supplemental reply.

Enclosure 3 identifies a regulatory commitment made in Enclosure 2 to modify the administrative controls regarding spent fuel pool cooling operations.

If you have any questions regarding 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 13th day of April, 2006.

Sincerely,

Original signed by:

William D. Crouch
Manager of Licensing
and Industry Affairs

Enclosures:

1. Revised Responses to RAIs IPSB-B.1 and IPSB-B.8
2. Supplements to Responses to RAIs SPLB-A.1, SPLB-A.2 and SPLB-A.3
3. Commitment Listing

cc (See page 3.)

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ENCLOSURE 1
TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 1, 2, AND 3

TECHNICAL SPECIFICATIONS (TS) CHANGE NOS. TS-418 AND TS-431 – REVISED RESPONSES TO NRC ROUND 2 REQUESTS FOR ADDITIONAL INFORMATION – IPSB-B.1 and IPSB-B.8

By letters dated December 19, 2005 (ADAMS Accession Nos. ML053560194 and ML053560186) TVA submitted responses to NRC Round 2 requests for additional information (RAIs) regarding TVA's applications for extended power uprate of BFN Unit 1 and BFN Units 2 and 3, respectively. As a result of discussions with the NRC staff, TVA is superseding its prior responses to RAIs IPSB-B.1 and IPSB-B.8 regarding external radiation doses due to direct radiation and skyshine. The responses to the subject RAIs are the same for all three BFN units.

NRC Request IPSB-B.1

Section 8.6, Normal Operations Off-Site Doses, of Enclosure 4 of the [June 25, 2004 and June 28, 2004 submittals for BFN Units 2 and 3 and BFN Unit 1, respectively] states that radiation from shine (offsite) is not presently a significant exposure pathway and is not significantly affected by EPU. This conclusion is based on the experience of earlier 5-percent power uprates for Units 2 and 3. Also, Section 8.2.2, Offsite Doses at Power Uprate Conditions, of the Environmental Report states that N-16 activity in the Turbine Building will increase linearly with EPU.

The magnitude of the N-16 source term in the Turbine Buildings is not a simple linear increase with reactor power. The equilibrium concentration of N-16 in the Turbine Building systems will be effected (an inverse exponential function) by the decreased decay resulting from the increased steam/feed flow between the reactor and the Turbine Building. Implementation of hydrogen injection water chemistry also increases N-16 concentrations in reactor steam independently of reactor power.

Provide the present nominal value for the skyshine external dose component (assuming all three units operating at current licensed power levels), the corresponding estimated dose component following EPU (assuming all three units operating at the requested power, and design basis steam activity, levels). Include all parameters (i.e., flow rates, system component dimensions, etc.) used in calculating these values and specify the calculational method used. Identify the limiting dose receptor (i.e., is the dose receptor a member of the public located offsite and, therefore, subject to the dose limits of 40 CFR Part 190) or a member of the public working onsite (subject to the dose limits of 20.1301)). Describe any increases in doses for onsite spaces (i.e., Administrative offices, guard stations, etc.) continuously or routinely occupied by plant visitors or staff.

TVA Reply to IPSB-B.1

A number of studies have been conducted at BFN to characterize the direct radiation and building/atmospheric scatter skyshine radiation fields associated with increased N-16 and C-15 production from hydrogen injection into the feedwater system for mitigation of intergrannular stress corrosion cracking (IGSCC) of vessel internals. Radiation levels onsite have been measured with thermoluminescent dosimeters (TLD), pressurized ionization chambers (PIC), and hyper-pure germanium detectors.

In 1997, while Units 2 and 3 were operating at original licensed thermal power (OLTP) (i.e., 3293 MWt), and prior to the injection of hydrogen in the feedwater system on either unit, GE Nuclear Energy performed extensive surveys of site radiation levels. A subsequent report, GE-NE-P7300044-01-01-00, "Browns Ferry Nuclear Power Station, Potential Dose Consequences Resulting From Implementation of Hydrogen Water Chemistry," provides dose rate projections through a number of shield wall thicknesses as a function of distance from BFN units under OLTP and normal water chemistry conditions. The projections were based on the output from the mathematical model provided in "BWR Turbine Equipment N-16 Radiation Shielding Studies," by D. R. Rogers, General Electric NEDO-20206, 1973, normalized to a PIC radiation measurement in line with the operating turbines at the north end of the electrical switchyard. This location was chosen because it is unaffected by other sources of radiation such as from radwaste processing/shipment or the condensate storage tanks. The report provided projections in-line with the turbines and normal to them. As the former were slightly more conservative (i.e., provided higher values), they were used for dose rate projections in occupied areas onsite. The dose rate projection curves in the report extend to 2000 feet from the turbine center line. The projection curves were then extrapolated in order to project site boundary doses. The dose rate at the nearest site boundary (i.e., 3850 feet) was projected to be 0.04 µR/h per unit under OLTP and normal water chemistry conditions. This equates to a total annual dose of approximately 1.1 mrem from all three units at OLTP.

Components on the turbine deck which contribute to the skyshine include: the piping to and from the high pressure turbine, the high pressure turbine, the crossover piping from the moisture separators to the low pressure turbines, the combined intercept valves and the low pressure turbines. According to the GE report (GE-NE-P7300044-01-01-00), the vast majority (~72%) of the skyshine emanates from N-16 and C-15 in the steam traversing the crossover piping. Based on this, the change in travel time of the steam to the midpoint of the crossover piping was calculated for OLTP, current licensed thermal power (CLTP) (i.e., 3458 MWt) and EPU (i.e., 3952 MWt) conditions. Steam travel times were calculated based on the steam flow rates for each of the power levels, piping layouts, and component configurations. The calculated travel times are shown in the following table:

Table IPSB-B.1-1
STEAM TRAVEL TIME

	Steam travel time to crossover piping mid-point (seconds)		
	OLTP (3,293 MWt)	**CLTP (3,458 MWt)**	**EPU (3,952 MWt)**
Unit 1	10.70	NA	8.93
Unit 2	10.49	10.12	8.63
Unit 3	10.47	10.10	8.62

The radiological decay of N-16 and C-15 was then calculated for those travel times, and a fractional increase in the radiation level was determined for each condition. The results for operation of all three BFN units were: an approximate 10% increase from OLTP to CLTP and an approximate 32% increase from CLTP to EPU.

By applying these increases to the GE dose projection for the OLTP condition, the annual dose to members of the public offsite and onsite were determined for CLTP and EPU under normal water chemistry conditions (i.e., no hydrogen injection). For calculation purposes, assuming three units at CLTP and EPU, the annual dose to a member of the public at the nearest terrestrial site boundary would be 1.2 and 1.5 mrem, respectively. Currently both Unit 2 and Unit 3 are operating with the addition of Noble Chem™ (platinum and rhodium) to the reactor coolant system and with reduced hydrogen injection in the feedwater for IGSCC mitigation. For three units at CLTP under reduced hydrogen injection, these values would be increased by approximately 25% to 1.4 and 1.9 mrem, respectively. The total dose to a member of the public includes effluent doses; however, these are negligible in comparison to the direct and skyshine radiation doses.

Therefore, the projected annual doses are well within the 25 mrem dose limit of 40 CFR 190 for an offsite member of the public.

The limiting dose receptors for members of the public would be those onsite (e.g., food vendors) because their work locations are nearer to the turbine building. The maximum annual dose to vendors would not likely exceed 18 mrem under EPU and Noble Chem™ water chemistry and reduced hydrogen injection conditions on all three units. Consequently, the 10 CFR 20.1301 annual dose limit of 100 mrem for a member of the public onsite would not be exceeded. Therefore, the projected annual dose to an onsite member of the public will be well within the dose limit.

Furthermore, following Unit 1 restart, with the reduction in restart workers and the re-location of a major portion of the site population into permanent structures farther from the turbine building, the increase in collective site dose from direct and skyshine radiation external to the plant structure is projected to be approximately 10 person-rem per year for three units at EPU conditions under Noble Chem™ water chemistry and reduced hydrogen injection.

NRC Request IPSB-B.8

Section 8.5.1, Normal Operations, of Enclosure 4 of the [June 25, 2004 and June 28, 2004 submittals for BFN Units 2 and 3 and BFN Unit 1, respectively] submittal states that, due to the conservative shielding design, the increase in radiation levels resulting from EPU will not affect the radiation zones for the various areas of the plant. This appears to be based on an assumed linear increase in radiation source term with power level. However, the increase in N-16 activity in the turbine building is an inverse exponential function with decay time, not a linear function of reactor power. Verify that the radiation zoning in all areas containing the steam and feed systems will be unaffected by EPU.

TVA Reply to IPSB-B.8

Under current licensed thermal power (CLTP) (i.e., 3458 MWt) conditions for BFN Units 2 and 3 with Noble Chem™ water chemistry and reduced hydrogen injection in the feedwater for IGSCC mitigation, all of the steam-affected areas, with the exception of the reactor feed pump turbine rooms, are locked high radiation areas (LHRA). This includes the reactor and turbine steam tunnels, moisture separator rooms, turbine rooms, high and low pressure heater rooms, condenser rooms, moisture separator drain pump and tank rooms, steam packing exhauster rooms, steam jet air ejector rooms, and hydrogen recombiner rooms. The reactor feed pump turbine rooms are posted as radiation areas at the entrances with smaller high radiation areas located inside the rooms enclosing the turbine and pump areas. The areas on the turbine roof over the turbine rooms are controlled as high radiation areas. Although BFN Unit 1 is currently not operating, nor is it currently licensed to operate above 3293 MWt, it is expected that its radiation levels will be consistent with the radiation levels of Units 2 and 3.

Under EPU conditions, the radiation levels are conservatively expected to increase by approximately 32% over the CLTP conditions. This is based on increased steam flow, reduction in steam travel time, and reduction in the radiological decay of N-16 and C-15. However this increase will not be enough to require changing the radiation area posting at the entrance to the rooms. In addition, the radiation zoning and posting outside the steam-affected area rooms are not expected to change due to EPU.

To ensure that proper postings are maintained, dose rates will be monitored in these environs during power ascension as part of the planned EPU testing.

ENCLOSURE 2
TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 1, 2, AND 3

TECHNICAL SPECIFICATIONS (TS) CHANGE NOS. TS-418 AND TS-431 – SUPPLEMENTS TO RESPONSES TO NRC ROUND 2 REQUESTS FOR ADDITIONAL INFORMATION – SPLB-A.1, SPLB-A.2 and SPLB-A.3

By letters dated December 19, 2005 (ADAMS Accession Nos. ML053560194 and ML053560186) TVA submitted responses to NRC Round 2 requests for additional information (RAIs) regarding TVA's applications for extended power uprate of BFN Unit 1 and BFN Units 2 and 3, respectively. As a result of discussions with the NRC staff, TVA is supplementing its prior responses to RAIs SPLB-A.1, SPLB-A.2 and SPLB-A.3 regarding fuel pool cooling. The responses to the subject RAIs are the same for all three BFN units.

NRC Request SPLB-A.1

Section 10.5.5 of the Updated Final Safety Analysis Report (UFSAR), Revision 17 dated August 30, 1999, revised the discussion from the UFSAR that was previously provided regarding the maximum SFP heat load for batch and full core offloads. In order to facilitate NRC review of the capability of the SFPCCS to perform its function for EPU conditions, provide a discussion on the safety-related systems required to maintain fuel pool cooling within design bases temperature limits.

NRC Request SPLB-A.2

For EPU conditions, explain how the SFP water temperature will be maintained below 150 degrees Fahrenheit (F) for the worst-case normal (batch) and full core offload scenarios assuming a loss of offsite power and (for the batch offload only) a concurrent single active failure considering all possible initial configurations that can exist. Include a description of the maximum decay heat load that will exist in the SFP for each case, how these heat loads were determined, such that they represent the worst-case conditions, and what the cooling capacity is for the systems that are credited, including how this determination was made. Also:

a. Describe any operator actions that are required, how long it will take to complete these actions, and how this determination was made; and

b. Describe the maximum core decay heat load that will exist at the onset of fuel movement, how this determination was made, how this heat load will be accommodated while also satisfying the SFP cooling requirements over the duration of the respective fuel offload scenarios, and including the situation where the SFP is isolated from the reactor vessel cavity.

NRC Request SPLB-A.3

Discuss how adequate SFP makeup capability is assured for EPU conditions in the unlikely event of a complete loss of SFP cooling capability, including how the maximum possible SFP boil-off rate compares with the assured makeup capability that exists, operator actions that must be taken, how long it will take to complete these actions and how this determination was made, and boron dilution considerations.

TVA's Supplemental Reply to SPLB-A.1, 2, and 3

TVA has previously provided information regarding the spent fuel pool cooling system at BFN and the effects of EPU in PUSAR Section 6.3 and in the December 19, 2005, reply to questions SPLB-A.1, SPLB-A.2, and SPLB-A.3. The following discussion is provided to clarify and provide supplemental information on the BFN spent fuel pool cooling system and is presented in the format (including numbering) of Attachment 2 to Matrix 5 of RS-001, "Review Standard for Extended Power Uprates," Revision 0, December 2003.

1. BACKGROUND

The BFN fuel pool cooling and cleanup systems for Units 1, 2, and 3 are described in UFSAR Section 10.5. The systems cool the fuel storage pools by transferring the spent fuel decay heat through heat exchangers to the reactor building closed cooling water (RBCCW) systems. The system for each fuel pool consists of two circulating pumps connected in parallel, two heat exchangers, one filter demineralizer subsystem, two skimmer surge tanks, and the required piping, valves, and instrumentation. Four filter demineralizers are provided including one spare filter demineralizer shared between the three units. The pumps circulate the pool water in a closed loop, taking suction from the surge tanks, circulating the water through the heat exchangers and filter demineralizer, and discharging it through diffusers at the bottom of the fuel pool and reactor well (as required during refueling operations). The water flows from the pool surface through skimmer weirs and scuppers (wave suppressers) to the surge tanks.

The heat exchangers in the residual heat removal (RHR) system can be used in conjunction with the fuel pool cooling and cleanup system to supplement pool cooling (supplemental fuel pool cooling). Normal makeup water for the fuel pool cooling system is transferred from the condensate storage tank to the skimmer surge tanks. A seismic Class I qualified source of makeup water is provided through the crosstie between the RHR system and fuel pool cooling system. If necessary, the intertie between the RHR service water (RHRSW) system and the RHR system can be utilized to admit raw water as makeup. Also, a standpipe and hose connection is provided on each of the two emergency equipment cooling water (EECW) system headers which provide two additional fuel pool water makeup sources.

Additionally, the auxiliary decay heat removal (ADHR) system provides another means to remove decay heat and residual heat from the spent fuel pool and reactor cavity of BFN Units 2 and 3 and is described in UFSAR Section 10.22. As part of restart activities for BFN Unit 1, the ADHR system will be extended to include the spent fuel pool and reactor cavity of BFN Unit 1. During operation of this system, it is aligned to only one unit at a time. The ADHR system consists of two cooling water loops. The primary cooling loop circulates spent fuel pool water entirely inside the Reactor Building and rejects heat to a secondary loop by means of a heat exchanger. The secondary loop transfers heat to the atmosphere outside the Reactor Building by means of evaporative cooling towers.

Spent fuel pool cooling, including supplemental fuel pool cooling and ADHR, are non-safety systems. To ensure adequate makeup under all normal and off normal conditions, the RHR/RHRSW connection provides a permanently installed seismic Class I qualified makeup water source for the spent fuel pool. This ensures that irradiated fuel is maintained submerged in water and that reestablishment of normal fuel pool water level is possible under all anticipated conditions. Two additional sources of spent fuel pool water makeup are provided via a standpipe and hose connection on each of the two EECW headers. Each hose is capable of supplying makeup water in sufficient quantity to maintain fuel pool water level under conditions of no fuel pool cooling.

2. <u>ACCEPTANCE CRITERIA</u>

The current design and operational basis for BFN spent fuel pool cooling system is as follows:

- Administrative controls are used to ensure that the fuel pool heat load does not exceed available cooling capacity.
- The capacity of the spent fuel pool cooling and the ADHR systems, considering seasonal cooling water temperatures and current heat exchanger conditions, are utilized to

maintain the fuel pool temperature at or below 125°F during normal refueling outages (average spent fuel batch discharged from the equilibrium fuel cycle).

- The RHR system can be operated in parallel with the spent fuel pool cooling system to maintain the fuel pool temperature less than the Technical Requirements Manual (TRM) limit of 150°F if a full core off load is performed. Plant instructions require that actions be taken well before exceeding this limit. The fuel pool temperature is normally maintained between 72°F and 125°F.
- To ensure adequate makeup under all normal and off normal conditions (i.e. fuel pool water boil off), the RHR/RHRSW crosstie provides a permanently installed seismic Class I qualified makeup water source for the spent fuel pool.
- Two additional sources of spent fuel pool water makeup are provided via a standpipe and hose connection on each of the two EECW headers. Each hose is capable of supplying makeup water in sufficient quantity to maintain fuel pool water level under conditions of no fuel pool cooling.

The design basis for the fuel pool cooling systems remains the same for the current and EPU conditions.

3. REVIEW PROCEDURES

3.1 Adequate SFP Cooling Capacity

To demonstrate adequate SFP cooling capacity, BFN performs both bounding and cycle-specific calculations. The bounding calculations have been reperformed for EPU conditions as described below in Section 3.1.1 to ensure that the acceptance criteria will continue to be met. Additionally, as described in Section 3.1.2, cycle-specific calculations are performed to assess cooling system capability to ensure that fuel pool heat load does not exceed available cooling capacity. These calculations demonstrate that the acceptance criteria described in Section 2 will continue to be met under EPU conditions.

As a result of EPU, the normal spent fuel pool heat load will be higher than the pre-EPU heat load. EPU will result in higher decay heat in the discharged bundles to the spent fuel pool as well as an increase in the number of discharged fuel bundles at the end of each cycle. The heat removal capability of the spent fuel pool cooling system, the ADHR system, or the supplemental fuel pool cooling mode of the RHR system are not affected by EPU. The evaluations for spent fuel pool cooling, as discussed below, include the effects from EPU operation and provide the results indicating that the design basis for the spent fuel pool will be maintained.

3.1.1 Bounding Calculation

Consistent with the BFN design basis, two cases were analyzed: 1. Partial core offload with operation of the spent fuel pool cooling system and ADHR system, and 2. Full core offload with operation of the spent fuel pool cooling system and RHR supplemental fuel pool cooling mode. In each case the initial fuel pool temperature was assumed to be 100°F.

1. Partial Core Offload

   The capacity of the fuel pool cooling system and the ADHR system to maintain the fuel pool temperature at or below 125°F during partial core offloads was evaluated for EPU conditions.

   The maximum decay heat loadings for the spent fuel pool were calculated using the ANSI/ANS 5.1-1979 Standard with two-sigma uncertainty. The heat load in the spent fuel pool is the sum of previous fuel offloads and the recent batch decay heats at the time of transfer. In this analysis, the offload consists of a batch of 332 fuel bundles offloaded to an almost full spent fuel pool. This batch size was chosen for analytical purposes; the actual batch size may vary.

   The spent fuel pool was assumed to be previously loaded with 2375 bundles allowing a reserve space for a full core offload (764 cells). The 2375 bundles were assumed to have been offloaded in eight batches, discharged at 24 month intervals. For this case, core offload begins 50 hours after reactor shutdown. Fuel transfer time was estimated based on a transfer rate of 14 bundles per hour to the fuel pool. These decay heat and offload time estimates establish the limiting case maximum heat loads.

   Cooling of the fuel pool conservatively assumes that only one heat exchanger/pump combination is available for each system. The heat exchanger effectiveness is based upon original design specifications including standard value fouling factors and tube plugging criteria. The evaluation only considers the mass of water in the fuel pool and assumes no circulation of water between the fuel pool and the cavity for the period of time that fuel pool gates are open while the fuel is being transferred to the pool.

The results of this evaluation show that the peak spent fuel pool temperature remains less than 125°F under EPU conditions.

Table 1
Partial Core Offload Evaluation Results for One Train Each of Spent Fuel Pool Cooling System and ADHR[1]

Conditions/Parameters	Value
Peak spent fuel pool temperature (°F)	99.1
Time to peak spent fuel pool temperature (hours)	80
Time to boil from loss of all cooling at peak temperature (hours)	14
Boil off rate (gpm)	48
[1] Assumes core offload begins 50 hours after reactor shutdown to allow for cooldown, vessel head removal, refueling cavity filling, and other refueling preparations.	

PUSAR Table 6-3 contains an additional case where a partial core offload was evaluated for one train each of the spent fuel pool cooling system and RHR supplemental fuel pool cooling mode. In that evaluation, the calculated peak spent fuel pool temperature of 124.9°F was less than 125°F.

Table 2

Partial Core Offload Evaluation Results for One Train Each of Spent Fuel Pool Cooling System and RHR Supplemental Fuel Pool Cooling Mode[1]

Conditions/Parameters	Value
Peak spent fuel pool temperature (°F)	124.9
Time to peak spent fuel pool temperature (hours)	130
Time to boil from loss of all cooling at peak temperature (hours)	13
Boil off rate (gpm)	42

[1] Assumes core offload begins 95 hours after reactor shutdown and includes 45 hours of invessel stay time because the RHR supplemental fuel pool cooling mode has less heat removal capacity than the ADHR.

2. <u>Full Core Offload</u>

The capacity of the spent fuel pool cooling system and the RHR supplemental fuel pool cooling mode to maintain the fuel pool temperature at or below 150°F during a full core off load is evaluated for EPU conditions.

The maximum decay heat loadings for the spent fuel pool were calculated using the ANSI/ANS 5.1-1979 Standard with two-sigma uncertainty. The heat load in the spent fuel pool is the sum of previous fuel offloads and the recent full core decay heats at the time of transfer. The pool is assumed to be previously loaded with 2707 bundles. The prior offload batches were assumed to be the same as the partial core offload case above with an additional batch of 332 fuel assemblies having been discharged from the reactor core, all of which has been cooled for an additional 24 months. (The partial offload batch size was chosen for analytical purposes; the actual may vary.) The initiation of fuel offloading was a minimum of 50 hours after plant shutdown based upon shutdown cooling requirements, head removal time and refueling preparation. Actual times were determined based on the calculated heat removal capacity of the cooling mode. For this case, core offload begins 165 hours after reactor shutdown and includes 115 hours of invessel stay time because the RHR supplemental fuel pool cooling mode has less heat removal capacity than the ADHR system. Fuel transfer time was estimated based on a transfer rate of 14 bundles per hour to the fuel pool. These decay heat and offload time estimates establish the limiting case maximum heat loads.

Cooling of the fuel pool conservatively assumes that only one heat exchanger/pump combination is available for each system. The heat exchanger effectiveness is based upon original design specifications including standard value fouling factors and tube plugging criteria. The evaluation only considers the mass of water in the fuel pool and assumes no circulation of water between the fuel pool and the cavity for the period of time that fuel pool gates are open while the fuel is being transferred to the pool.

The results of this evaluation show that the peak spent fuel pool temperature remains less than 150°F under EPU conditions.

Table 3

Full Core Offload Evaluation Results for One Train Each of Spent Fuel Pool Cooling System and RHR Supplemental Fuel Pool Cooling Mode[1]

Conditions/Parameters	Value
Peak spent fuel pool temperature (°F)	149.8
Time to peak spent fuel pool temperature (hours)	229
Time to boil from loss of all cooling at peak temperature (hours)	4
Boil off rate (gpm)	80
[1] Assumes core offload begins 165 hours after reactor shutdown and includes 115 hours of invessel stay time because the RHR supplemental fuel pool cooling mode has less heat removal capacity than the ADHR.	

PUSAR Table 6-3 contains an additional case where a full core offload was evaluated for one train each of the spent fuel pool cooling system and ADHR system. In that evaluation, the calculated peak spent fuel pool temperature of 121.5°F was also less than 150°F.

Table 4

Full Core Offload Evaluation Results for One Train Each of Spent Fuel Pool Cooling System and ADHR[1]

Conditions/Parameters	Value
Peak spent fuel pool temperature (°F)	121.5
Time to peak spent fuel pool temperature (hours)	109
Time to boil from loss of all cooling at peak temperature (hours)	5
Boil off rate (gpm)	104
[1] Assumes core offload begins 50 hours after reactor shutdown to allow for cooldown, vessel head removal, refueling cavity filling, and other refueling preparations.	

3.1.2 Cycle-Specific Calculation

Unloading the reactor core and the associated increase in fuel pool heat load is a controlled evolution. Administrative controls are used to ensure that the fuel pool heat load does not exceed available cooling capacity, such that the fuel pool gates are not closed until the decay heat load is less than or equal to the fuel pool cooling heat exchanger capacity. Performance of the fuel pool cooling systems is predicted prior to each refueling outage as part of the Outage Risk Assessment Review (ORAM) process.

In addition to the following discussion, BFN is taking additional actions to further augment procedures pertaining to the cycle specific administrative controls. Procedure changes will be generated (1) to define and control the generation of cycle-specific fuel pool heat load calculations, and (2) to control the installation of the fuel pool gates based on the calculated fuel pool heat load.

Cycle-specific analysis conditions:

(1) Predicted decay heat for both the spent fuel pool and reactor core are determined by utilizing a TVA code (DHEAT) that complies with the methods of ANSI/ANS 5.1. The history of previous fuel discharges is used as input into the decay heat load determination for the spent fuel pool. The decay heat results are best-estimate values and are provided for a range of decay times that may be needed for the spent fuel pool evaluations.

(2) Cooling system heat removal is calculated utilizing a spreadsheet based on heat balances of the affected systems. Fuel pool cooling capacity of the systems is based upon inlet cooling temperatures, system flow rates, trains in service, and heat exchanger performance values.

(3) As described in (2) above, heat removal capabilities are determined for each of the BFN cooling trains, including the normal spent fuel pool cooling system, the ADHR system, and the supplemental fuel pool cooling mode of RHR.

(4) The limiting parameter for heat load and heat removal capability is the insertion of the fuel pool gates following core offload. When the fuel pool gates are removed and spent fuel movement begins, additional

cooling is provided by the shutdown cooling system that provides decay heat removal directly to the reactor vessel. Evaluations of the spent fuel pool temperature following discharge of the partial core offload are performed based on cooling system configurations to ensure that the spent fuel pool temperature can be maintained without the additional heat removal capacity of the shutdown cooling system.

(5) (As discussed with the NRC staff on April 13, 2006, TVA will address this item in a supplemental reply.)

(6) Administrative controls are provided as part of ORAM to ensure that appropriate controls are provided for shutdown safety. These controls ensure proper assessment of key shutdown areas (i.e., reactivity control, shutdown cooling, AC power, fuel pool cooling, etc.). Spent fuel pool cooling assessments are performed prior to the outage and updated during the outage to ensure appropriate controls are maintained for the safe operation of spent fuel pool cooling.

3.2 Adequate Make-Up Supply

The evaluations described in Sections 3.1.1.1 and 3.1.1.2 above are used to determine the time to boil for make-up capability. These evaluations assume only one train of each cooling system is in operation to determine the peak spent fuel pool temperature. At the time of peak spent fuel pool temperature, it is assumed that all spent fuel pool cooling is lost. Based on decay heat, the time to reach boiling conditions is then calculated. The results are provided in Tables 1 through 4 above.

The minimum time to reach boiling is four hours based on the case presented in Table 3. This case involves a full core offload and assumed loss of all cooling at the peak spent fuel pool temperature of 149.8°F. The associated boil off rate is 80 gpm.

The maximum boil off rate is 104 gpm based on the case presented in Table 4. This case involves a full core offload and assumed loss of all cooling at the peak spent fuel pool temperature of 121.5°F. The associated time to reach boiling is five hours.

For BFN the RHR/RHRSW crosstie provides a permanently installed seismic Class I qualified makeup water source for the spent fuel pool. This supply can be aligned within the minimum four hours calculated above and can supply greater than 150 gpm to the spent fuel pool.

Two additional sources of spent fuel pool water makeup are provided via a standpipe and hose connection on each of the two EECW headers. Each hose is capable of supplying makeup water at 150 gpm to the spent fuel pool within the minimum four hours calculated above.

ENCLOSURE 3

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT (BFN)

UNITS 1, 2, AND 3

TECHNICAL SPECIFICATIONS (TS) CHANGE NOS. TS-418 AND TS-431 – COMMITMENT LISTING

Prior to implementing EPU, procedure changes will be generated (1) to define and control the generation of cycle-specific fuel pool heat load calculations, and (2) to control the installation of the fuel pool gates based on the calculated fuel pool heat load.