May 17, 2004

BFN-TS-405

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

Gentlemen:

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

BROWNS FERRY NUCLEAR PLANT (BFN) - UNITS 1, 2, AND 3 - RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION (RAI) AND UNIT 1 ANALYSIS RESULTS RELATED TO TECHNICAL SPECIFICATIONS (TS) CHANGE NO. TS-405 - ALTERNATIVE SOURCE TERM (AST) (TAC NOS. MB5733, MB5734, MB5735)

This letter provides additional information requested by NRC in support of TS-405 and the results of the Unit 1 AST analysis for the Loss-of-Coolant Accident (LOCA), the Main Steam Line Break Accident, and the Control Rod Drop Accident. TS-405, which was submitted on July 31, 2002, requested a license amendment and TS changes for a full scope application of AST methodology for BFN Units 1, 2, and 3.

NRC provided the RAI by letter dated April 16, 2004. Enclosure 1 provides TVA's response to each of the staff's questions.

Through the BFN Unit 1 restart effort, TVA has identified chloride bearing cable in the Units 2 and 3 primary containments not previously reported. Consequently, TVA updated portions of the BFN Safety Assessment concerning pH control during a LOCA provided in TVA's July 31, 2002, TS change. The updated portions are provided in Enclosure 3.
On December 9, 2002, TVA responded to NRC questions concerning TS-405. TVA's response to Request 14 included design input data which has changed as a result of the discovery of the not previously reported chloride bearing cable. For completeness and accuracy, TVA is providing a revised response to NRC Request 14 in Enclosure 2.

In the July 31, 2002 letter, TVA also provided a Units 1, 2, and 3 license amendment for AST. Because BFN shares a common refueling floor, the radiological dose analysis for the refueling floor was valid for all three units. The analysis for the LOCA, the Main Steam Line Break Accident, and the Control Rod Drop Accident were performed for Units 2 and 3 but not Unit 1. The Unit 1 analysis has been completed and Enclosure 3 provides the results of these analyses. The Unit 1 inputs for the LOCA analysis were different than used for Units 2 and 3 therefore, separate analyses were performed for Unit 1. The analysis indicates that the Unit 1 DBA-LOCA offsite doses are larger than Units 2 and 3. The Units 2 and 3 control room doses for a DBA-LOCA are larger than Unit 1. The bounding results are provided in the revised Safety Assessment in Enclosure 3. Revisions to the Safety Assessment are identified by a line drawn in the left margin.

TVA has reviewed the changes in Enclosure 2 and the results provided in Enclosure 3 and determined that the information in these do not alter the No Significant Hazards Consideration previously published. TVA requests approval of the license amendment TS-405 by July 31, 2004, and the changes to the TSs made effective within 120 days of NRC approval.

There are no regulatory commitments contained in this letter. If you have any questions about this, please telephone me at (256) 729-2636.

Pursuant to 28 U. S. C. § 1746 (1994), I declare under penalty of perjury that the foregoing is true and correct. Executed on May 17, 2004.

Sincerely,

Original signed by:

T. E. Abney
Manager of Licensing
and Industry Affairs

Enclosures
1. Response To the April 16, 2004, Request For Additional Information (RAI) Relating To Technical Specifications Change No. TS-405 Alternative Source Term (AST)
2. BFN AST Safety Assessment Replacement Pages
3. Units 1, 2, and 3 Evaluation Results

cc: See page 4:

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ENCLOSURE 1

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT
UNITS 1, 2, AND 3

RESPONSE TO THE APRIL 16, 2004, REQUEST FOR ADDITIONAL INFORMATION (RAI) RELATING TO TECHNICAL SPECIFICATIONS CHANGE No. TS-405 ALTERNATIVE SOURCE TERM (AST)

On July 11, 2003 (Reference 1), TVA submitted and Exemption Request to 10 CFR 50 Appendix A, General Design Criteria – 41 to allow the use of the Standby Liquid Control (SLC) system to limit the consequences of a Loss-of Cooling Accident (LOCA). The SLC system operation is relied upon to inject sodium pentaborate thus minimizing the re-evolution of iodine into the containment atmosphere, which reduces the dose consequences. That letter addressed most of the questions in the April 16, 2004, RAI. Also, some of the questions have been addressed in other pervious correspondence with NRC by letters dated December 9, 2002, (Reference 2) and July 31, 2002 (Reference 3).

In the cases where the question has been previously addressed by TVA, the applicable portion of the letter is provided with our reply to the question.

<u>NRC Request 1</u>

Please identify whether the SLC system is classified as a safety-related system as defined in Title 10, *Code of Federal Regulations* (10 CFR), Section 50.2 and whether the SLC system satisfies the regulatory requirements for a safety related system. If the SLC system is not classified as safety-related, please provide the information requested in Items 1.1 to 1.5 below, to show that the SLC system is comparable to a system classified as safety-related. If any item is answered in the negative, explain why the SLC system should be found acceptable for pH control agent injection.

<u>NRC Request 1.1</u>

Is the SLC system provided with standby AC power supplemented by the emergency diesel generators?

<u>TVA Response 1.1</u>

The Standby Liquid Control (SLC) system is classified as a special safety system as defined in the BFN Updated Final Safety Analysis Report (UFSAR). The SLC system is provided with standby AC power supplemented by the emergency diesel generators. The SLC system supplies are discussed on page E1-5 of the

July 11, 2003 (Reference 2) letter. The following is an excerpt from the referenced correspondence.

> The SLC System is required to be operable in the event of a station power failure. Therefore, the pumps, valves, and controls are powered from the standby AC power supplies. The pumps and valves are powered and controlled from separate buses and circuits so that a single power failure will not prevent system operation. Separate 250-V DC, battery backed, distribution panels powered from their respective 480-V shutdown board powers each pump control circuit and the injection valves. The injection valves are continuously monitored and alarm in the main control room if either circuit opens.

NRC Request 1.2

Is the SLC system seismically qualified in accordance with Regulatory Guide 1.29 and Appendix A to 10 CFR Part 100 (or equivalent used for original licensing)?

TVA Response 1.2

The SLC system components required for reactivity control and the AST function are Seismic Class 1. The seismic qualification of the SLC system is discussed on page E1-26 of the December 9, 2002 (Reference 1) letter. The following is an excerpt from the referenced correspondence.

> Using the SLC system to control pH in the suppression pool following a postulated LOCA with fuel damage constitutes a new SLC system function which is consistent with its use in special events. The system has qualities that ensure its reliability, such as:
>
> - Seismic Class 1 design of components required for reactivity control and new suppression pool pH control functions.

NRC Request 1.3

Is the SLC system incorporated into the plant's Inservice Inspection or Testing Program (e.g., 10 CFR 50.55a) based upon the American Society of Mechanical Engineers Boiler and Pressure Vessel Code?

TVA Response 1.3

The required SLC system components are included in the American Society of Mechanical Engineers (ASME) inservice inspection program. These requirements are discussed on page E1-26 of the December 9, 2002, letter and page E1-6 of the July 11, 2003 letter. The following is an excerpt from the referenced correspondence.

From the December 9, 2002 letter:

Using the SLC system to control pH in the suppression pool following a postulated LOCA with fuel damage constitutes a new SLC system function which is consistent with its use in special events. The system has qualities that ensure its reliability, such as:

- Subject to American Society of Mechanical Engineers Section XI In Service Inspection requirements.

From the July 11, 2003 letter:

The containment isolation check valves are American Society of Mechanical Engineers (ASME) Code Class 2 valves, and are subject to ASME Section XI Inservice Testing and 10 CFR 50 Appendix J Local Leak rate Testing (LLRT) Program. In accordance with these programs, these check valves are inspected and tested during scheduled refueling outages. TS require a system flow test to the vessel, which demonstrates the operability of the integrated system at least once an operating cycle (24 months). A review of the maintenance history does not indicate any failures of these valves to open or close.

The SLC system pump discharge check valves are identical to the containment isolation check valves. These are exercised quarterly under the ASME Section XI program and are inspected on a regular basis. The inspections have not identified any indication of wear or other unusual degradation.

NRC Request 1.4

Is the SLC system incorporated into the plant's Maintenance Rule program consistent with 10 CFR 50.65?

TVA Response 1.4

The SLC system is within the scope of the BFN 10 CFR 50.65 Maintenance Rule program. Discussion of the Maintenance Rule program as it relates to the SLC system is on page E1-9 of the July 11, 2003 letter. The following is an excerpt from the referenced correspondence.

The SLC system is within the scope of the BFN 10 CFR 50.65 Maintenance Rule program. The BFN probabilistic Safety Analysis (PSA) establishes the system performance criteria, balancing unavailability and reliability for risk significant critical structures systems, and components. TVA's maintenance rule program requires that each SLC pump flow path (subsystem) shall maintain an unavailability factor less than or equal to 1.347E-02 on a rolling 24-month interval. The observed unavailability factors for the SLC system are considerably less than the required unavailability factor.

The maintenance rule program monitors and trends all SLC system unavailability that could affect the flow path availability. Both A and B pumps, including the containment isolation check valves are included in the program requirements. The total recorded unplanned unavailability for the flow paths is as follows:

	Pump 2A[1]	Pump 2B[1]	Pump 3A[2]	Pump 3B[2]
Total Unplanned Unavailable Hours	12.9	0.0	9.1	1.4

1) Since June 1994
2) Since Unit 3 restart, November 1995

Unit 1 is in long-term layup and no SLC system data is available. The Unit 1 system is identical to Units 2 and 3. Therefore, TVA anticipates that the Unit 1 SLC performance will be consistent with Units 2 and 3.

The inclusion of the SLC system into the 10 CFR 50.65 Maintenance Rule and the confirmation of acceptable performance provides a continued assurance of the availability for performance of the AST function.

NRC Request 1.5

Does the SLC system meet 10 CFR 50.49 and Appendix A to 10 CFR 50 (General Design Criteria No. 4, or equivalent used for original licensing)?

TVA Response 1.5

As a special safety system, the SLC system is not currently subject to the requirements of 10 CFR 50.49 and Appendix A to 10 CFR 50 for the LOCA. However; as part of the implementation of TS-405, TVA is in the process of qualifying the SLC system components to the post-LOCA conditions they will be subjected to during the performance of the AST function.

NRC Request 2

Please describe the proposed changes to plant procedures that implement SLC sodium pentaborate injection as a pH control additive. In addition, address Items 2.1 to 2.5 below in your response. If any item is answered in the negative, explain why the SLC system should be found acceptable for pH control additive injection.

NRC Request 2.1

Are the SLC injection steps part of a safety-related plant procedure?

TVA Response 2.1

TVA currently has SLC activation steps in the plant procedures associated with the anticipated transient without scram (ATWS) event. As previously discussed on page E1-6 of the July 11, 2003 letter, the steps necessary to inject SLC system into the vessel during a postulated LOCA that results in fuel damage are no different than during an ATWS. The following is an excerpt from the referenced correspondence.

> To mitigate ATWS, SLC must be initiated within a few minutes. In contrast, AST analysis credits SLC system initiation within two hours post-LOCA. As discussed in the July 31, 2002, license amendment request, TVA will revise plant procedures to require the initiation of the SLC system based on indication of fuel failure (high radiation in the primary containment).
>
> The new SLC function does not involve any change to the operator steps needed to initiate SLC injection. The timing requirements for operator response are considerably relaxed for AST compared to ATWS analysis requirements since the AST analysis assumes the system initiation is within two hours of the event.

NRC Request 2.2

Are the entry conditions for the SLC injection procedure steps symptoms of imminent or actual core damage?

TVA Response 2.2

The entry conditions for SLC injection will be based on symptoms of actual core damage. AST implementation involves changing the appropriate BFN Alarm Response Procedure to require SLC system injection based on indication of high drywell radiation. The entry conditions for SLC injection for AST implementation are discussed on page 16 of the July 31, 2002, Enclosure 4 Safety Assessment. The following is an excerpt from the referenced correspondence.

> Initiation of the SLC System **following fuel damage** to control suppression pool pH is a new operator action during a DBA LOCA response. High radiation indicative of fuel failure would be sensed by two radiation monitors in the drywell and two radiation monitors in the pressure suppression chamber. Upon reaching a high radiation level, the "Drywell/Suppr Chamber Radiation High" annunciator on Panel 9-7 in the main control room would alert the operator to the fuel damage. The Alarm Response Procedure (ARP) will direct the operator to initiate SLC System injection based on the high radiation level.

NRC Request 2.3

Does the instrumentation cited in the procedure entry conditions meet the quality requirements for a Type E variable as defined in RG 1.97 Tables 1 and 2?

TVA Response 2.3

Indication of fuel damage during a postulated LOCA is provided by two high range containment area radiation monitors. Detailed discussion on these instruments is provided on page E1-25 of the December 9, 2002, letter. The following is an excerpt from the referenced correspondence.

> Two high range containment area radiation monitors (RM-90-272A and RM-90-273A) provide independent and redundant indication, recording, and alarm functions in the CR. These radiation monitors are listed in TS 3.3.3.1, Post Accident Monitoring (PAM) Instrumentation, and are Category 1/class 1E equipment designed to meet RG 1.97 (Reference 14). Digital printout, an alarm printout, and a control room annunciator alarm are provided. These monitors are used in BFN's Radiological Emergency Planning program procedures to estimate core damage, hence, use in an AST capacity is consistent with the current use.

NRC Request 2.4

Have plant personnel received initial and periodic refresher training in the SLC injection procedure?

TVA Response 2.4

Part of operator training and periodic retraining includes training on the current SLC system ATWS function. As discussed on pages 16 and 17 of Enclosure 4 in the July 31, 2002 letter, TVA will provide training on the new pH control function during a design basis LOCA prior to AST implementation and during the normal re-qualification training. The following is an excerpt from the referenced correspondence.

> Initiation of the SLC System will be accomplished from the main control room with a simple keylock switch manipulation. This switch is located on control room panel 9-5 and actuation of this switch is the only action necessary to initiate injection of the sodium pentaborate into the reactor vessel. The new SLC System function to control suppression pool pH does not involve any change to the actions needed to be performed to initiate SLC system injection. Indication of proper SLC System operation is provided in the control room as described in UFSAR Section 3.8.

During this postulated event, plant operators will be responding to the event as directed by the plant Emergency Operating Instructions (EOI). Adequate time is available for SLC System initiation during these events. Immediate initiation of the SLC System is not vital since the analysis allows for two hours before initiation. Operators are familiar with operation of the SLC System due to previous training for Anticipated Transients Without Scram (ATWS) events. Training on this new operator action will also be provided to the operators.

NRC Request 2.5

Have other plant procedures (e.g., Emergency Response Guidelines/Severe Accident Guidelines) that call for termination of SLC as a reactivity control measure been appropriately revised to prevent blocking of SLC injection as pH control measure?

TVA Response 2.5

See TVA's response to NRC Request 2.2.

NRC Request 3

Please provide a description of the analysis assumptions, inputs, methods, and results that show that a sufficient quantity of sodium pentaborate can be injected to raise and maintain the suppression pool greater than pH 7 within 24 hours of the start of the event. (See also Position 2 of Appendix A to RG 1.183.) In your response, please discuss the adequacy of recirculation of suppression pool liquid via emergency core cooling system through the reactor vessel and the break location and back to the suppression pool in meeting the transport and mixing assumptions in the chemical analyses. Assume a large-break LOCA.

TVA Response to Request 3

A description of the analysis that shows that a sufficient quantity of sodium pentaborate can be injected to raise and maintain the suppression pool greater than pH 7 for 30 days within 24 hours of the start of a postulated LOCA with fuel damage is in Enclosure 2.

TVA's mixing evaluation conservatively includes the initial maximum liquid volume of the suppression pool (approximately 982,900 gal), the liquid volume of the reactor vessel and recirculation system volume (approximately 109,300 gal), and the volume of the SLC system injection credited for pH control (approximately 4,000 gal), a total volume of approximately 1,096,200 gallons.

The BFN LOCA analysis assumes an electrical failure in one division. This result in one loop of Core Spray (CS) available to the vessel at a flow rate of 5,600 gpm, and one loop of Residual Heat Removal (RHR) available for suppression pool cooling at a flow rate of 13,000 gpm. TVA's LOCA analysis assumes suppression pool cooling is underway within 10 minutes of the start of the event.

The initiation of the injection of sodium pentaborate by the SLC system is within 2 hours of the event and the injection is completed in less than 2 hours of initiation. This is consistent with the event timing specified in RG 1.183. The CS ECCS injection flow delivered to the reactor vessel mixes with the sodium pentaborate and spills from the reactor vessel through the recirculation system line break, flushing sodium pentaborate to the suppression pool. A complete turn over of the reactor vessel and recirculation system volume by the available CS system occurs approximately three times an hour. Since both RHR and CS take suction from the suppression pool a turnover of the suppression pool volume occurs approximately once every hour.

Based on the above, adequate transport of the sodium pentaborate to the suppression pool as well as suppression pool recirculation mixing will occur prior to the time credit is needed for the buffering effect of the sodium pentaborate for pH control.

NRC Request 4

Please show that the SLC system has suitable redundancy in components and features to assure that for onsite or offsite electric power operation, its safety function of injecting sodium pentaborate for the purpose of suppression pool pH control can be accomplished assuming a single failure. For this purpose, the check value is considered an active device since the check valve must open to inject sodium pentaborate. If the SLC system can not be considered redundant with respect to its active components, the licensee should implement one of the three options described below, providing the information specified for that option for staff review.

4.1 Option 1 Show acceptable quality and reliability of the nonredundant active components and/or compensatory actions in the event of failure of the nonredundant active components. If you choose this option, provide the following information to justify the lack of redundancy of active components in the SLC system:

NRC Request 4.1.1

Identify the nonredundant active components in the SLC system and provide their make, manufacturer, and model number.

TVA Response 4.1.1

The nonredundant active components in the SLC system, the containment isolation valves and the main control room common start switch, are described on pages E1-6 and E1-8 in the July 11, 2003 letter. The following is an excerpt from the referenced correspondence.

> Check Valves
>
> The containment isolation check valves are stainless steel Velan 1½-inch Bolted Bonnet Piston Check Valves (model W7234B13MS), mounted horizontally in the injection line. For an ATWS event, the containment isolation check valves are designed to open against full reactor pressure. For the AST function, the system operating requirements are reduced since the reactor pressure is much lower.
>
> Common Start Switch
>
> The SLC system is actuated by a five-position switch located in the main control room. The switch is a General Electric (GE) type SB-1 nine-stage rotary cam-operated switch, and is used in both safety and non-safety related applications at BFN. The nine individual stages are stacked onto a common shaft and mechanically tied together with two bolts threaded into the front support. Each stage has two contacts. The entire contact assembly is enclosed in a metal cover that provides physical protection for the switch contacts. This switch is used thoughtout the industry, and is of simple construction with few parts vulnerable to failure. The typical mechanical service life for this switch is estimated to be approximately one million cycles.

NRC Request 4.1.2

Provide the design-basis conditions for the component and the environmental and seismic conditions under which the component may be required to operate during a design-basis accident. Environmental conditions include design-basis pressure, temperature, relative humidity and radiation fields.

TVA Response 4.1.2

All SLC system equipment used for pH control following a LOCA and subject to environmental conditions per 10 CFR 50.49 is either physically located within the secondary containment in the Reactor Building or the Main Control Room.

The SLC system pH control function is complete within 6 hours post-LOCA. Environmental analysis reflects a 6-hour post-LOCA temperature of 96° F in the SLC equipment area. The area pressure and humidity are not affected post-LOCA. The maximum abnormal pressure and relative humidity are 14.4 psig and 90 percent during the LOCA accident. Hence, the requirements of 10 CFR 50.49 are not applicable for these parameters.

The current EPU design basis 40-year normal radiation for the SLC system general area is 8.41E4 Rads and the 60-year normal dose is 1.26E5 Rads. These values are based on a specific piping surface (of another system) contact dose value in the room of 240 mR/hr. The location of this value in the room is remote from the SLC equipment. Applying the EPU scaling factors to the original licensing thermal power radiation doses, the integrated accident airborne dose after 6 hours in the SLC system general area, assuming infinite cloud geometry, is 5.18E+03 Rads (gamma) and 3.30E+04 Rads (beta). The design basis radiation analyses and SLC equipment are being evaluated in order to complete equipment qualification in accordance with the requirements of the BFN Environmental Qualification Program (See response to RAI 1.5).

The SLC power and control circuitry cables are routed from their terminal locations through Reactor Building areas to power sources and control locations. The SLC pump motor power cables are routed in a combination of conduit and tray in the Reactor Building from the pumps to respective 480V shutdown boards. The control cables are powered from 250V DC control circuits for the respective 480V shutdown boards and are run in a combination of conduit and tray in the Reactor Building from the local panel to the 480V shutdown board and then on the control room panels. The squib valves are fired by 250V DC control power, with the controls being in the main control room and the power coming from the 250V DC boards which are in turn powered from the 480V shutdown boards. Their cables are run in common trays from the main control room into the reactor building and on to the SLC System equipment room with their routing being in separate conduits.

The cables and cable routing of the SLC system are being identified along with their respective Reactor Building LOCA environments in order to complete cable evaluation and qualification to the requirements of the BFN Environmental Qualification Program (See response to RAI 1.5).

NRC Request 4.1.3

Indicate whether the component was purchased in accordance with Appendix B to 10 CFR Part 50. If the component was not purchased in accordance with Appendix B, provide information on the quality standards under which it was purchased.

TVA Reply 4.1.3

Equipment that provides the 10 CFR 50.62 ATWS function is subject to the requirements of the TVA's Augmented Quality Program. The Quality Assurance program as is relates to the SLC system components is discussed on page E1-10 of the July 11, 2003 letter. The following is an excerpt from the referenced correspondence.

> Equipment that provides the 10 CFR 50.62 ATWS function is a special safety system and, as such, is required by in the Quality Assurance Program to meet quality related standards subject to the requirements of the Augmented Quality Program. The TVA Nuclear Quality Assurance Plan defines "quality-related" as:
>
> "…a term which encompasses quality assurance program requirements that describe activities which affect structures, systems, and components. These requirements provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public. In addition to safety related structures, systems, components, and activities, the term "quality related" encompasses the broad class of plant features covered (not necessarily explicitly) in the General Design Criteria of 10 CFR 50, Appendix A, that contribute in an important way to the safe operation and protection of the public in all phases and aspects of facility operation (i.e., normal operation and transient control as well as accident mitigation)."
>
> Although the controls of the Augmented Quality Program are not as stringent as for equipment classified as safety-related, the significance of the "quality related" classification ensures a consistent means to control quality features, including matters such as procurement of replacement parts and control of maintenance activities. The SLC system parts are procured like for like, by part number and description. The procured part is traceable to the requisition contract and traceability is maintained for the life of the plant. Receipt inspection ensures part requested, meets the procurement requirements (part number and description), verifies no shipping damage, establishes a shelf life, and establishes any special preventative maintenance activities for storage of part prior to installation in plant. Maintenance activities are second party verified by an individual qualified to perform the task.
>
> The rigorous controls imposed by the Quality Assurance Program provide more than adequate quality control elements to ensure SLC system component reliability for the required special safety function under the ATWS rule and for the performance of the AST function.

NRC Request 4.1.4

Provide the performance history of the component both at the licensee's facility and in industry databases such as Equipment Performance Information and Exchange System and Nuclear Plant Reliability Data System.

TVA Response 4.1.4

The performance history of the nonredundant components, the SLC system injection check valves and the common start switch, are discussed on pages E1-7 and E1-8 of the July 11, 2003 letter. The following is an excerpt from the referenced correspondence.

> Check Valves
>
> NUREG/CR 5944 (9/93), "A Characterization of Check Valve Degradation and Failure Experience in the Nuclear Power Industry," documented a review and evaluation of check valve failures. The review found that the overall failure rate of the study for all check valves was 0.00996 per year. The failure rate of the check valves ≤ 2 inches was 0.00706 per year. In the distribution of failures, the restricted flow motion and failed closed modes were responsible for only 7 percent of the valve failures. In the ≤ 2 inches size group, the fraction of stuck closed failures was approximately 0.15 resulting in a failed closed failure rate of 0.001059 per year.
>
> The Nuclear Industry Check Valve Group (NIC) established a centralized check valve failure/reliability database based in part on the existing Institute of Nuclear Power Operations Nuclear Plant Reliability Data System data. A sort of the NIC database for Velan lift/piston check valve failures was performed to identify any reported failures and the associated failure modes. Sixty-one instances of Velan 1½-inch lift/piston check valve failures were identified. Of the failures, only two failures were identified as stuck closed. Both stuck closed valves were carbon steel valves being operated in a wet steam environment. The SLC system containment isolation check valves are stainless steel valves. During the SLC system functional testing demineralized water is pumped through the valves. Therefore, the operating environment for the failed valves is not consistent with the BFN SLC system.
>
> A failure summary report from the Institute of Nuclear Power Operations Equipment Performance and Information Exchange (EPIX 4.0) database shows no instances of the same model valve as BFN's check valves (Velan W7234B13MS) to fail stuck closed. Additionally, an industry experience review performed for the BFN

condition-monitoring program did not identify any failures of this type for this valve to open. Based on our review, there have been no failures of this type identified in the nuclear operating history for the manufacturer and model lift piston check valves in the BFN SLC system.

In summary, industry data indicates check valves 2 inches and less are very reliable. Further, Velan 1½-inch check valves have experienced only 2 stuck closed failures. The model check valves in the BFN SLC system have not experienced a stuck closed failure. Based on this operating experience, the stuck closed failure of these valves in the common discharge line is highly unlikely. Therefore, the underlying purpose of GDC-41 is met by providing a highly reliable means of controlling fission products with the current design.

Common Start Switch

This switch is used throughout the industry, and is of simple construction with few parts vulnerable to failure. The typical mechanical service life for this switch is estimated to be approximately one million cycles.

A review of EPIX exchange database identified three GE SB-1 switch failures in the industry. Contact corrosion attributed to two of the failures. The remaining failure was a result of sticking or binding. These failed switches were not in a main control room environment. The BFN switches are located in a temperature and humidity controlled environment in the main control room and not subject to contact corrosion.

NRC Request 4.1.5

Provide a description of the component's inspection and testing program, including standards, frequency, and acceptance criteria.

TVA Response 4.1.5

The functional testing requirements of the SLC system injection check valves and common start switch are implemented through the BFN TSs. The system testing is discussed on page E1-11 of the July 11, 2003 letter. The following is an excerpt from the referenced correspondence.

Once every operating cycle (24 months), system functional testing verifies one subsystem's pump discharge relief valve setpoint and the pumping capacity and the ability to inject into the reactor vessel. The functional test alternates each subsystem being tested. During the functional test, operation of the control circuits, indicators, and the alarm annunciator operation are verified.

NRC Request 4.1.6

Indicate potential compensating actions that could be taken within an acceptable time period to address the failure of the component. An example of a compensating action might be the ability to jumper a switch in the control room to overcome its failure. In your response consider the availability of compensating actions and the likelihood of successful injection of the sodium pentaborate when nonredundant active components fail to perform their intended functions.

TVA Response to 4.1.6

No additional compensatory actions are considered necessary to ensure injection of SLC through the required flow path. TVA's rationale is provided in the summary on page E1-14 of the July 11, 2003 letter. The following is an excerpt from the referenced correspondence.

> In the BFN AST analysis, the SLC system is credited for limiting the radiological consequences following a design basis LOCA involving significant fuel damage. The use of the SLC system to provide a buffering solution for the suppression pool following the postulated design basis LOCA credits SLC injection as a new function. The adequacy of the SLC system to perform this function is supported by the system design and physical configuration. System maintenance practices and high system reliability is supported by required TS surveillance testing and the Inservice Test Program, and the requirements of the BFN Quality Assurance Program. Although the SLC system does not strictly meet all single failure requirements, the SLC system is suitable for the AST function.
>
> SLC system initiation (and injection) for AST will continue to be a manual operator action requiring the same operator steps necessary for an ATWS event except that the timing requirements are relaxed. In the AST analyses, the SLC system initiation is based on plant alarm response to conditions indicating postulated fuel damage (high containment radiation). The initiation conditions are consistent with the conditions expected within the primary containment following the release of fission products.
>
> The PSA values assigned to SLC indicate the system is highly reliable. The system is very simple to operate. One switch in the main control room controls the SLC system operation. Redundant pumps and explosive valves ensure that at least one subsystem of SLC will operate when required. Operating and control power for the system is backed by emergency power sources. Review of the single failure

aspects of this system determined the portions of the SLC system that do not meet single failure requirements are highly reliable and are unlikely to fail. The single line into containment and the reactor vessel contain two series check valves. TVA's review of industry operating experience for these valves determined that they are very reliable. TVA's review of industry experience for the GE SB-1 main control room switch indicates it is very reliable.

Additionally, TVA has evaluated the consequences of SLC failing to control pH in the suppression pool and determined the resulting dose consequences would remain within the regulatory limits.

4.2 Option 2 Provide for an alternative success path for injecting chemicals into the suppression pool. If you chose this option, provide the following information.

4.2.1 Provide a description of the alternative injection path, its capabilities for performing the pH control function, and its quality characteristics.

4.2.2 Do the components which make up the alternative path meet the same quality characteristics required of the SLC system as described in Items 1.1 to 1.5, 2 and 3 above?

4.2.3 Does the alternate injection path require actions to be taken in areas outside the control room? How accessible will these areas be? What additional personnel would be required?

TVA Response 4.2

Based on TVA's reply to Option 1, no alternative success path for injection into the suppression pool will be required.

4.3 Option 3 Show that 10 CFR 50.67 dose criteria are met even if pH is not controlled. If you chose this option, demonstrate through analyses that the projected accident doses will continue to meet the criteria of 10 CFR 50.67 assuming that the suppression pool pH is not controlled. The dissolution of cesium iodide and its re-evolution from the suppression pool as elemental iodine must be evaluated by a suitably conservative methodology. The analysis of iodine speciation should be provided for staff review. The analysis documentation should include a detailed description and justification of the analysis assumptions, inputs, methods, and results. The resulting iodine speciation should be incorporated into the dose analyses. The calculation may take credit for the mitigating capabilities of other equipment, for example the standby gas treatment system, if such equipment would be available. A description of the dose analysis assumptions, inputs, methods, and results should be provided. Licensees proposing this approach should recognize that this option will incur longer staff review times and will likely involve fee-billable support from national laboratories

TVA response to Request 4.3

In support of the Exemption from General Design Criteria-41, TVA performed a dose consequence evaluation assuming no SLC operation. The evaluation included the dose consequences for the Exclusion Area Boundary, the Low Population Zone and the Control Room. A summary of the evaluation and methodology is provided in Enclosure 2 of TVA's July 11, 2003, letter. The dose estimates remained below regulatory limits even with no credit for SLC operation.

NRC Request 5

The Updated Final Safety Analysis Report for BFN Section 1.5.1.6, "Nuclear Design Criteria," has requirements on the secondary containment in Items 13, 14, 15 and automatic responses in Item 3, and control room shielding in Item 23. These requirements may be impacted by the proposed Technical Specification (TS) changes, which relax requirements on secondary containment operability and isolation functions. Provide information on BFN's compliance with these nuclear design criteria and any proposed changes to these criteria which will be made.

TVA Response to Request 5

BFN UFSAR Section 1.5 provides the principal architectural and engineering criteria, which defines the broad frame of reference within which the plant designed. Detailed information concerning the safety design basis unique to the requirements of these various structures and systems described in Section 1.5 are provided in Chapters 2 through 13 of the BFN UFSAR.

The refueling accident analysis in TVA's TS-405 does not depend on any of the features described in BFN UFSAR Section 1.5.1.6, Items 13, 14, 15, 3 and/or 23. The principal safety objective considered when these architectural and engineering features, under which the plant was designed and constructed continues to be met. The analysis indicates continued acceptable safety results following a postulated refueling accident without secondary containment operability and isolation functions.

Nuclear Design Criteria 13: The secondary containment shall be designed to act as a radioactive material barrier under the same conditions that require the primary containment to act as a radioactive barrier.

The secondary containment will continue to act as a radioactive material barrier for the conditions that require the primary containment to act as a radioactive material barrier. The proposed TS changes continue to require that secondary containment be maintained anytime primary containment is required for operation if any one of the three BFN units. Although BFN is a three unit facility, the secondary containment is common to all three units. The only time the BFN would not maintain secondary containment is if all three units are in Modes 4 and 5. The results of TVA's analysis provided in Table 3-3 from the TVA Safety Assessment (Reference 3) indicates that if a postulated refueling accident were to occur during the time secondary containment may not be available, the on site and off site doses are well within the regulatory limit.

<u>Nuclear Design Criteria 14</u>: The secondary containment shall be designed to act as a radioactive material barrier, if required, whenever the primary containment is open for expected operational purposes.

The secondary containment design will continue to act as a radioactive material barrier, "if required," whenever the primary containment is open for expected operational purposes. As stated above, our analyses indicate that secondary containment is not required during a postulated refueling accident. As such, implementation of AST will not change this requirement.

<u>Nuclear Design Criteria 15</u>: The primary and secondary containments, in conjunction with other engineered safeguards, shall act to prevent the radiological effects of accidents resulting in the release of radioactive material to the containment volumes from exceeding the guideline values of applicable regulations.

As shown in Table 3-3 of TVA's Safety Assessment (Reference 3), TVA's analysis indicates that the results of the refueling accident are well within the regulatory guidelines even though a release is assumed through the refuel building ventilation with no credit taken for reactor building holdup or dilution.

<u>Nuclear Design Criteria 3</u>: Where positive, precise action is immediately required in response to accidents, such actions shall be automatic and shall require no decision of manipulation of controls by plant personnel.

See TVA's discussion on Nuclear Design Criteria 15.

<u>Nuclear Design Criteria 23</u>: The control room shall be shielded against radiation so that occupancy under accident conditions is possible.

TVA has not altered the shielding capability afforded by the design of the main control room for a refueling accident. As shown in Table 3-3 of

Enclosure 2 in the December 9, 2002, letter to NRC our analysis indicates that the dose to the operator from a refueling accident is well within the regulatory guidelines.

NRC Request 6

As described in TVA's submittals on BFN AST, there appears to be no intent to restore isolation to the secondary containment or to stop venting the secondary containment building in the event of a fuel-handling accident. Other licensees have committed to TS changes or administrative controls that would require restoration of containment and termination of venting after a fuel-handing accident. Please provide information on actions, plans, or commitments that BFN intends to make or implement in the event of a fuel handling accident or other radiological release in an open secondary containment.

TVA Response to Request 6

TVA will put in place administrative controls that will require restoration of the secondary containment and termination of secondary containment venting following a postulated fuel handling accident.

References:

1. TVA letter to the NRC dated July 11, 2003, "Browns Ferry Nuclear Plant (BFN) - Units 1, 2, And 3 - Exemption Request From The Requirements Of 10 CFR 50 Appendix A General Design Criteria (GDC)-41 In Support Of Technical Specifications Change (TS-405) - Alternative Source Term (AST)."

2. TVA letter to the NRC dated December 9, 2002, "Browns Ferry Nuclear Plant (BFN) - Units 1, 2, And 3 - Response To Request For Additional Information (RAI) Relating To Technical Specifications (TS) Change No. TS-405 - Alternative Source Term (AST)."

3. TVA letter to the NRC Dated July 31, 2002, "Browns Ferry Nuclear Plant (BFN) – Units 1, 2, and 3 - License Amendment – Alternative Source Term."

ENCLOSURE 2

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT
UNITS 1, 2, AND 3

RESPONSE TO THE APRIL 16, 2004, REQUEST FOR ADDITIONAL INFORMATION (RAI) RELATING TO TECHNICAL SPECIFICATIONS CHANGE No. TS-405 ALTERNATIVE SOURCE TERM (AST)

BFN AST SAFETY ASSESSMENT REPLACEMENT PAGES

This enclosure provides replacement pages for the December 9, 2002 reply RAI. Changes in the information previously submitted are shown by a line drawn in the right margin.

Table 2-16 Suppression Pool pH Control Inputs			
Input/Assumption	**Value**		
Maximum Suppression Pool Volume	131,400 ft^3		
Containment Free Volume	278,400 ft^3		
Reactor Coolant System Inventory	1.226E 6 lbm		
Sodium Pentaborate Injectable Volume	4000 gal		
SLC ($Na_2O*5B_2O_3*10H_2O$) injected	8 weight percent		
Sodium Pentaborate Enrichment	62.9 mole% B10		
Initial Suppression Pool pH	5.3		
Average suppression pool temperature	132°F		
Drywell Cable Data			
Hypalon Jacketed Cables	*Unit 1*	*Unit 2*	*Unit 3*
Mass of Jacket	3703 lbm	868 lbm	868 lbm
Average Cable Outside Diameter	0.89 inches	0.89 inches	0.89 inches
Average Cable Jacket Thickness	72 mils	72 mils	72 mils
Percent of Cable in Conduit	30%	50%	50%
Percent of Cable in Trays	70%	50%	50%
Polyvinyl Chloride Jacketed Cables	*Unit 1*	*Unit 2*	*Unit 3*
Mass of Jacket	NA	1389 lbm	1389 lbm
Average Cable Outside Diameter	NA	0.89 inches	0.89 inches
Average Cable Jacket Thickness	NA	72 mils	72 mils
Percent of Cable in Conduit	NA	30%	30%
Percent of Cable in Trays	NA	70%	70%
Neoprene Jacketed Cables	*Unit 1*	*Unit 2*	*Unit 3*
Mass of Jacket	1492 lbm	1492 lbm	1492 lbm
Average Cable Outside Diameter	0.73 inches	0.73 inches	0.73 inches
Average Cable Jacket Thickness	72 mils	72 mils	72 mils
Percent of Cable in Conduit	0%	0%	0%
Percent of Cable in Trays	50%	50%	50%
Halar Jacketed Cables[1]	*Unit 1*	*Unit 2*	*Unit 3*
Mass of Jacket	NA	155.4 lbm	NA
Average Cable Outside Diameter	NA	0.236 inches	NA
Average Cable Jacket Thickness	NA	25 mils	NA

Table 2-16 Suppression Pool pH Control Inputs			
Input/Assumption	Value		
Percent of Cable in Conduit	NA	0%	NA
Percent of Cable in Trays	NA	0%	NA
Conduit Material	Aluminum		
Conduit wall thickness	0.1 inch		
Conduit air gap	0.25 inch		

[1] Temporary cable installed in Unit 2 and planned to be removed during a future outage.

Revised reply to the Response 14 from Enclosure 1 of the December 9, 2002 Letter.

Background

The BFN pH calculation methodology used the Polestar STARpH 1.04 software (Reference 10). STARpH was developed and is maintained under Polestar's 10 CFR 50, Appendix B Quality Assurance Program, and has been validated against several experiments and more detailed pH models.

Purpose of pH Calculation

The BFN pH calculation determines the suppression pool post-accident pH vs. time out to 30 days using the 8% solution of sodium pentaborate from the SLCS tank as a buffer.

Methodology

- Calculate the [HNO_3] concentration in the suppression pool water as function of time post-LOCA using the Radiolysis of Water model of the STARpH 1.04 code
- Calculate the [HCl] concentration in the water pool as a function of time using the Radiolysis of Cable model of the STARpH 1.04 code
- Manually calculate the [H^+] concentration added to the pool as a function of time from the results of the above calculations
- Determine the time-averaged post-LOCA temperature of the suppression pool
- Determine the dissociation constant of the sodium pentaborate buffer, using the time-averaged post-LOCA temperature of the suppression pool
- Determine the starting pH of the sodium pentaborate buffered solution.
- Calculate the boron concentration corresponding to the design input volume of SLCS (4000 gal) with a solution of 8 weight % sodium pentaborate
- Calculate the suppression pool pH as a function of time using the Add Acid model of the STARpH 1.04 code

Design Input Data

1. Reactor power = 4031 MWth (102 % of 3952 MWth)
2. Maximum volume of water in suppression pool = 131,400 ft^3

3. RCS inventory = 1.226E6 lbm*
4. Pool initial pH = 5.3
5. Average Suppression Pool Temperature = 132° F
6. Fraction of aerosol depositing in pool = 0.79
7. Fission product inventory and source term, same as for DBA-LOCA dose analysis
8. Mass of Cable jacket = See the following table
9. Thickness of Cable jacket = See the following table
10. Air gap in conduit = 0.25 inch, see Assumption below
11. Conduit wall thickness = 0.1 inch, see Assumption below
12. Conduit material = aluminum, see Assumption below
13. Drywell free volume = 159,000 ft^3
14. Minimum torus free volume = 119,400 ft^3
15. Volume of sodium pentaborate in SLCS = 4000 gal.
16. Sodium pentaborate concentration in SLCS = 8 weight %
17. Density of SLCS containing 8 weight % sodium pentaborate = 8.64 lbm/gal.
18. Chemical formula for sodium pentaborate = $Na_2O \bullet 5B_2O_3 \bullet 10H_2O$
19. Boron enrichment in sodium pentaborate is 62.9 mole % B^{10}
20. Drywell coating = 28,780 ft^2 epoxy coating
21. Torus coating = 34,014 ft^2 epoxy coating

Assumption: The conduit surrounding a portion of the electrical cabling is aluminum of 0.1 inch wall thickness and it has an air gap of 0.25 inches.

Justification: The shielding from conduit increases with the density of the conduit material and the thickness of the conduit and is inversely proportional to the thickness of the air gap between the cable and the conduit. This is based on evaluations of shielding effect of conduit. The assumption of aluminum of 0.1 in thickness and an air gap of 0.25 inch provide a shielding factor of about 20.

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* Table 2-16 indicates 1.226E-06 lbm. A revised Table 2-16 is in Enclosure 2.

BFN CABLE DATA			
	Unit 1	**Unit 2**	**Unit 3**
Hypalon Jacketed Cables:			
Total Footage, ft.	57,193	23,680	23,680
Total Jacket Weight, lbs	3,703	868	868
Average Cable OD, in.	0.89	0.89	0.89
Jacket Thickness, mils	72	72	72
% in Conduit, %	30	50	50
% in Tray, %	70	50	50
% in Free Air, %	0	0	0
PVC Jacketed Cables:			
Total Footage, ft.	0	17,402	17,402
Total Jacket Weight, lbs	N/A	1,389	1,389
Average Cable OD, in.	N/A	0.89	0.89
Jacket Thickness, mils	N/A	72	72
% in Conduit, %	N/A	30	30
% in Tray, %	N/A	70	70
% in Free Air, %	N/A	0	0
Neoprene Jacketed Cables:			
Total Footage, ft.	16,650	16,650	16,650
Total Jacket Weight, lbs	1,492	1,492	1,492
Average Cable OD, in.	0.73	0.73	0.73
Jacket Thickness, mils	72	72	72
% in Conduit, %	0	0	0
% in Tray, %	50	50	50
% in Free Air, %	50	50	50
Halar Jacketed Cables:			
Total Footage, ft.	N/A	14,000	N/A
Total Jacket Weight, lbs	N/A	155.4	N/A
Average Cable OD, in.	N/A	0.236	N/A
Jacket Thickness, mils	N/A	25	N/A
% in Conduit, %	N/A	0	N/A
% in Tray, %	N/A	0	N/A
% in Free Air, %	N/A	100	N/A

Calculation of HCl, HNO_3, and $[H^+]$ Added to Pool (mole/L)

Time	Net $[OH^-]$	[HCl] (U1)	HCl] (U2)	[HCl] (U3)	$[HNO_3]$	$[H^+]$ Added (U1)	$[H^+]$ Added (U2)	$[H^+]$ Added (U3)
1h	1.19E-4	9.79E-06	1.34E-05	1.25E-05	5.87E-6	1.57E-05	1.93E-05	1.84E-05
2h	1.17E-4	1.85E-05	2.54E-05	2.36E-05	8.06E-6	2.65E-05	3.34E-05	3.17E-05
5h	1.12E-4	3.91E-05	5.38E-05	5.00E-05	1.26E-5	5.17E-05	6.63E-05	6.26E-05
12h	1.05E-4	7.36E-05	1.01E-04	9.40E-05	2.00E-5	9.36E-05	1.21E-04	1.14E-04
1d	9.49E-5	1.17E-04	1.60E-04	1.49E-04	2.98E-5	1.47E-04	1.90E-04	1.79E-04
3d	6.71E-5	2.35E-04	3.22E-04	3.00E-04	5.75E-5	2.92E-04	3.80E-04	3.58E-04
10d	1.57E-5	4.08E-04	5.61E-04	5.22E-04	1.09E-4	5.17E-04	6.70E-04	6.31E-04
20d	(1.72E-5)	4.75E-04	6.53E-04	6.07E-04	1.42E-4	6.17E-04	7.95E-04	7.49E-04
30d	(3.86E-5)	4.98E-04	6.84E-04	6.36E-04	1.63E-4	6.61E-04	8.47E-04	8.00E-04

The data in the table are calculated such that the "Net $[OH^-]$" includes the net effects of both fission product CsOH and the formation of HNO_3. A positive "Net $[OH^-]$" indicates (on its own) a basic solution. The "$[H^+]$ Added" is the sum of the HNO_3 and the HCl. "$[H^+]$ Added" does not include the favorable effects of CsOH. These data do not yet consider the effects of the sodium pentaborate buffer.

Required Sodium Pentaborate

The calculation of the amount of sodium pentaborate necessary to maintain pH above 7 for 30 days after the accident was performed using the STARpH code. This calculation does the following:

- Input the concentration of buffer (in this case, borate buffer) in the pool and the dissociation constant for the buffer
- Establish the starting pH of the buffered solution. Suggested starting pH values are given in the StarpH documentation for a variety of situations and buffer materials commonly encountered in reactor analysis.
- In STARpH, two buffers are permitted to be acting simultaneously in the calculation; various borate and phosphate buffers are included in StarpH
- The BFN has only one buffer (borate from the sodium pentaborate solution in the SLCS)
- Input the total strong acid (i.e., mol/L of HNO_3 and HCL)
- Calculate the final pH

ENCLOSURE 3

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT
UNITS 1, 2, AND 3

RESPONSE TO THE APRIL 16, 2004, REQUEST FOR ADDITIONAL INFORMATION (RAI) RELATING TO TECHNICAL SPECIFICATIONS CHANGE No. TS-405 ALTERNATIVE SOURCE TERM (AST)

UNITS 1, 2, AND 3 EVALUATION RESULTS

2. EVALUATION

2.1 Scope

2.1.1 Accident Radiological Consequence Analyses

The DBA accident analyses documented in Chapter 14 of the BFN UFSAR (Reference 4) that could potentially result in control room and offsite doses were addressed using methods and input assumptions consistent with the AST. The following DBAs were addressed:

- LOCA, UFSAR Section 14.6.3
- Main Steam Line Break Accident, UFSAR Section 14.6.5
- Refueling Accident, UFSAR Section 14.6.4
- Control Rod Drop Accident, UFSAR Section 14.6.2

The analysis was performed per RG 1.183. The results were evaluated to confirm compliance with the acceptance criteria presented in 10 CFR 50.67 and GDC 19 of 10 CFR 50, Appendix A. Computer codes used in the DBA analyses results are listed in Table 2-1.

The AST control room dose analyses are applicable for all three unit control rooms. The Unit 1 and 2 control rooms are shared in a common room with Unit 1 at one end and Unit 2 at the other. The Unit 3 control room, though separated from the Units 1 and 2 control room, is part of the same control bay habitability zone.

The inputs used in the Main Steam Line Break Accident, Refueling Accident, and Control Rod Drop Accident analyses are bounding for Units 1, 2, and 3; therefore, the results determined for these events are applicable for all three units. The inputs used in the LOCA analysis are different for Unit 1 and Units 2 & 3; therefore, a separate analysis is performed and the bounding results for Units 1, 2, and 3 are provided.

2.1.2 Suppression Pool pH Control

A calculation was performed to evaluate the suppression pool pH in the event of a DBA LOCA. The objective of the analysis was to demonstrate that the suppression pool pH remains at or above 7.0, thus ensuring that the particulate iodine (cesium iodide - CsI) deposited into the suppression pool during this event does not re-evolve and become airborne as elemental iodine. The analysis credits the pH buffering effect of sodium pentaborate introduced into the suppression pool post-LOCA by SLC operation to maintain the pH above 7.0.

The ORIGEN code (Reference 6) was used to calculate plant-specific fission product inventories which bound the effect of two-year fuel cycles, power operation at EPU conditions (4031 MWt (102% of 3952 MWt)), and using current and anticipated fuel designs. The fission product inventory for General Electric (GE)-14, Framatome Atrium-10 fuel, and Framatome Blended Low Enriched Uranium (BLEU) fuel designs were evaluated. Bounding values of fission product activity were determined for each radionuclide in the DBA radiological analyses. Fission product activities were calculated for immediately after shutdown and 24 hours following shutdown. The values are shown in Table 2-2.

The RADTRAD computer code Version 3.02(a) (Reference 7) was used for the DBA dose calculations. The computer code STARDOSE (Reference 8) was used to check the RADTRAD results. The RADTRAD and STARDOSE programs are radiological consequence analysis codes used to determine post-accident doses at offsite and control room locations. The STARDOSE code is the proprietary property of Polestar Applied Technology, Inc., and the NRC has previously reviewed results obtained from the application of this code.

The existing UFSAR X/Q values were developed prior to and used in support of the license amendment request (References 9 and 10) for increased main steam isolation valve (MSIV) leakage rate limits. Control room X/Q values for the base of the stack releases were calculated using the computer code ARCON96 (Reference 11). For sites such as BFN with control room ventilation intakes that are close to the base of tall stacks, ARCON96 under predicts the X/Q values for top of stack releases; therefore, top of stack releases to the control room intakes were evaluated using the methods of Regulatory Guides 1.111 (Reference 12) and 1.145 (Reference 13). The X/Q values associated with top of stack, base of stack, and turbine building roof ventilator releases were reviewed by the NRC in the Safety Evaluation for Amendments 263 and 223 for BFN Units 2 and 3, respectively (Reference 14). For Units 2 and 3 the X/Q values associated with the turbine building roof ventilator are more conservative than the turbine building exhaust release X/Q values; therefore, the turbine building roof ventilator X/Q values were used in the DBA LOCA analyses for Unit 2 and 3. For Unit 1, X/Q values associated with the turbine building exhaust release are more conservative than the turbine building roof ventilator X/Q values; therefore, the turbine building exhaust X/Q values were used in the DBA LOCA analyses for Unit 1. The existing X/Q values applicable to the time periods, distances, and geometric relationships are shown in Tables 2-3 through 2-7. Existing values for X/Q were used for AST radiological dose analyses except for the establishment of a new control room X/Q value associated with an instantaneous ground level puff release for the case of a main steam line break accident (see Section 2.2.3).

The post-LOCA shine dose to personnel in the control room includes the radiation shine from the secondary containment airborne activity and gamma dose from Core Spray System piping, which is in close proximity to the control building. Evaluations were performed of the existing TID-14844 analysis to determine applicable shine dose values for AST. For radiation from the Core Spray System piping, a comparison of gamma radiation plots from the suppression pool water was performed for high energy photons to determine similarity of shapes for the TID-14844 source term and the AST source term.

For the secondary containment airborne shine dose, a shine dose multiplier for AST airborne radioiodines was developed to enable direct comparison of the TID-14844 and the AST shine dose. To support this comparison, the activity for TID-14844 was increased to account for the increase in power level. The resulting comparison of several key nuclides found that the AST I-131 and I-133 activities in the reactor building are approximately a factor of 3 lower at 1.3 hours and 5 hours and, a factor of 30 lower at 24 hours compared to the TID-14844 levels at the same times. Considering the highest multiplier for the AST radionuclides (used to account for the activity other than iodine, especially for cesium) for 1 to 8 hours and at 24 hours, the effective iodine activity airborne in the reactor building for AST would be about the same before 8 hours and about a factor of 10 lower at 24 hours compared to TID-14844. For noble gases, the AST activities are about a factor of two lower than the TID-14844 source term at two hours, and by 24 hours, they are about the same.

The evaluation established that the integrated gamma dose from Core Spray System piping is slightly higher than previous over the 720 hours duration of the accident for the AST. However, only about 25 percent of the total 720 hour control room shine dose is due to the Core Spray System piping contribution. The control room shine dose from airborne activity in the secondary containment will be substantially reduced for the AST as compared to the TID-14844 source term. Therefore, the existing integrated control room shine dose, even if increased by the power ratio of EPU, is acceptable for a combination of EPU operation and AST application. This evaluation was checked using the MicroShield code, Version 5.03 (Reference 15). MicroShield is a point kernel integration code used for general purpose gamma shielding analysis. MicroShield has been used in safety-related applications by many nuclear plants in the United States. In this BFN application, it has been used as a means for design verification as an independent analysis.

For a DBA LOCA in Unit 1, the secondary containment airborne shine dose is affected by the difference in the reactor building effective mixing free volume. Due to the reduced mixing volume and no change in the SGT flow rate, the reduced holdup of activity in the reactor building reduces the control room shine dose for Unit 1.

For the main steam line break accident, radiation shine from the turbine building was conservatively handled assuming all released inventory is in the turbine building for two hours. Radiation shine from the airborne activity having escaped the turbine building is handled explicitly by the TVA computer code COROD. The calculation incorporates the control building dimensions and concrete roof (2.25 ft thick) in conjunction with the main steam line break accident released radioisotopes in a cloud above the control building.

release at the base of the stack. This amount of leakage is within the bounds of procedural controls.

The reactor building effective mixing free volume for Units 2 and 3 consists of 50% of the combined free volume of the affected reactor building and refuel floor. For Unit 1, the refueling floor portion of the effective mixing free volume is further reduced by 50% to account for the proximity of the SGT suction to the Unit 1 equipment hatch.

Since the main steam lines and the main condenser are seismically-rugged, and are assumed to remain intact, the MSIV leakage eventually collects in the main condenser (except for a small portion that is assured to bypass the main condenser). The LOCA analysis also assumes that one of the four inboard MSIVs fails to close (this postulated single failure results in the worst case dose consequences). Therefore, three of the steam lines have a closed space between the inboard and outboard MSIVs. The piping volume between the outboard MSIVs and the assorted valving downstream (i.e., main turbine stop valves, main turbine bypass valves, reactor feed pump high pressure steam stop valves, etc.) also comprises a large, closed space. In each of the three steam lines that are fully isolated, a well mixed control volume is defined in the space between the closed MSIVs as well as in the space downstream of the outboard MSIVs.

Only the control volumes in the horizontal portions of this main steam piping are credited in the analyses for activity disposition. The space down stream of the MSIV in the faulted steam line (the one with only the outboard MSIV closed) is credited with an isolated control volume only in the space from the outboard MSIV to the point where the drain line pathway to the main condenser connects to the steam line. This volume is consistent with others in that it is made up of horizontal piping also.

For conservatism, a maximum MSIV leakage per line of 100 scfh is assumed to exist in the faulted line. One of the fully isolated lines is assumed to leak at 50 scfh, while the other two are assumed to be leak-tight. This set of assumptions minimizes credit for retention in the steam lines.

The pressure in the space between the closed MSIVs is assumed to be that of the containment, but the temperature is assumed to be the normal operating conditions of the steam line. In the steam line outboard of the MSIVs, the pressure is assumed to be atmospheric, the temperature is also assumed to be the normal operating. The condenser is assumed to be at standard conditions. MSIV leakage at the test pressure is converted into volumetric flow rates based upon post-LOCA drywell temperature and pressure.

The MSIV leakage from the main condenser is assumed to be released directly to the environment as a turbine building release with no credit for turbine building hold-up.

The control room would automatically isolate and the CREV is automatically initiated at the onset of the accident due to high drywell

Table 2-3
X/Q Values for Radiological Dose Calculations
Top of Stack Releases

(LOCA and Control Rod Drop Accident)

Time Period	Control Room (sec/m^3)		EAB[(2)]	LPZ
	Unit 1 Intake	Unit 3 Intake	(sec/m^3)	(sec/m^3)
Fumigation	3.40E-5	*	2.35E-5[1]	1.26E-5
0-2 hrs	**	1.41E-7	1.19E-6[1]	1.13E-6
2-8 hrs	**	4.50E-8	—	5.75E-7
8-24 hrs	**	2.54E-8	—	4.10E-7
1-4 days	**	7.36E-9	—	1.97E-7
4-30 days	**	1.24E-9	—	6.88E-8

1 These values were incorrectly listed in Reference 14; however, the correct values were used as the basis of Reference 14.
2 Maximum EAB TEDE for any 2 hour period.
* Bounded by the Unit 1 Intake
** Bounded by the Unit 3 Intake

Table 2-4
X/Q Values for Radiological Dose Calculations
Base of Stack Releases

(LOCA and Control Rod Drop Accident)

Time Period	Control Room (sec/m^3)		EAB[(2)]	LPZ
	Unit 1 Intake	Unit 3 Intake	(sec/m^3)	(sec/m^3)
0-2 hrs	2.00E-4	*	2.62E-4	1.31E-4
2-8 hrs	1.28E-4	*	—	6.61E-5
8-24 hrs	5.72E-5	*	—	4.69E-5
1-4 days	4.05E-5	*	—	2.23E-5[1]
4-30 days	3.09E-5	*	—	7.96E-6

1 Typo in Reference 14; same as value for turbine building release.
2 Maximum EAB TEDE for any 2 hour period
* Bounded by the Unit 1 Intake

Table 2-5 **X/Q Values for Radiological Dose Calculations** **Refueling Vent Releases** (Refueling Accident Only)				
Time Period	Control Room (sec/m^3)		EAB	LPZ
	Unit 1 Intake	Unit 3 Intake	(sec/m^3)	(sec/m^3)
0-2 hrs	4.60E-4	*	2.62E-4	1.31E-4

*Bounded by the Unit 1 Intake

Table 2-6 **X/Q Values for Radiological Dose Calculations** **Turbine Building Exhaust Release** (Main Steam Line Break Accident - EAB/LPZ; Post-LOCA MSIV Leakage - Unit 1 Only)				
Time Period	Control Room (sec/m^3)		EAB	LPZ
	Unit 1 Intake	Unit 3 Intake	(sec/m^3)	(sec/m^3)
0-2 hrs	3.22E-4	*	2.62E-4	1.31E-4
2-8 hrs	2.77E-4	*	—	6.61E-5
8-24 hrs	1.31E-4	*	—	4.69E-5
1-4 days	7.91E-5	*	—	2.23E-51
4-30 days	6.10E-5	*	—	7.96E-6

*Bounded by the Unit 1 Intake.

Table 2-7 X/Q Values for Radiological Dose Calculations Turbine Building Roof Ventilator Releases (Post LOCA MSIV Leakage - Units 2/3 Only)				
Time Period	Control Room (sec/m^3)		EAB(1)	LPZ
	Unit 1 Intake	Unit 3 Intake	(sec/m^3)	(sec/m^3)
0-2 hrs	*	2.17E-4	2.62E-4	1.31E-4
2-8 hrs	*	1.64E-4	—	6.61E-5
8-24 hrs	*	7.89E-5	—	4.69E-5
1-4 days	*	4.33E-5	—	2.23E-5
4-30 days	*	3.35E-5	—	7.96E-6

*Bounded by the Unit 3 Intake

1 Maximum EAB TEDE for any 2 hour period

<table>
<tr><th colspan="3">Table 2-12
LOCA Inputs</th></tr>
<tr><th>Input/Assumption</th><th colspan="2">Value</th></tr>
<tr><td>MSIV Leak Rate at test pressure of 25 psig</td><td colspan="2">150 scfh total
100 scfh maximum for one line</td></tr>
<tr><td>Leakage at base of stack (stack bypass)</td><td colspan="2">10 scfm</td></tr>
<tr><td>MSIV Leakage that Bypasses Main Condenser</td><td colspan="2">0.5%
(percentage of total MSIV leakage)</td></tr>
<tr><td>CAD vent rate</td><td colspan="2">139 scfm for 24 hrs
@ 10 days, 20 days, 29 days</td></tr>
<tr><td>Volumes</td><td colspan="2"></td></tr>
<tr><td>Drywell Airspace</td><td colspan="2">159,000 ft^3
(Min value used for dose calculation)</td></tr>
<tr><td>Torus Airspace</td><td colspan="2">119,400 ft^3 (Minimum)</td></tr>
<tr><td>Suppression Pool</td><td colspan="2">121,500 ft^3 (Minimum)</td></tr>
<tr><td>Reactor Building Effective Mixing Free Volume</td><td>Unit 1
1,311,209 ft^3</td><td>Units 2 or 3
1,931,502 ft^3</td></tr>
<tr><td>Stack Room</td><td colspan="2">69,120 ft^3
(50% of this value used due to incomplete mixing)</td></tr>
<tr><td>High Pressure Turbine</td><td colspan="2">568.6 ft^3
(No credit taken)</td></tr>
<tr><td>Low Pressure Turbine</td><td colspan="2">51,000 ft^3
(No credit taken)</td></tr>
<tr><td>Removal Inputs</td><td colspan="2"></td></tr>
<tr><td>Drywell Natural Deposition</td><td colspan="2"><u>Particulate</u>: Power's Model, 10 th percentile values(conservative compared to SRP 6.5.2 λ_w.
<u>Elemental</u>: Same as particulate.</td></tr>
<tr><td>Drywell Accident Conditions (maximum)</td><td colspan="2">P = 48.5 psig,
T = 295.2 Degrees F</td></tr>
<tr><td>Surface Area for Elemental Iodine Deposition in Drywell</td><td colspan="2">3409 m^2</td></tr>
</table>

Table 2-15 Control Rod Drop Accident Inputs	
Input/Assumption	**Value**
Activity released from the condenser	Noble Gas 100% Iodine 10% Br 1% Cs, Rb 1% Te Group 1% Ba, Sr 1% Noble Mtls 1% Ce Group 1% La Group 1%

Table 2-16 Suppression Pool pH Control Inputs			
Input/Assumption	**Value**		
Maximum Suppression Pool Volume	131,400 ft^3		
Containment Free Volume	278,400 ft^3		
Reactor Coolant System Inventory	1.226E 6 lbm		
Sodium Pentaborate Injectable Volume	4000 gal		
SLC ($Na_2O*5B_2O_3*10H_2O$) injected	8 weight percent		
Sodium Pentaborate Enrichment	62.9 mole% B10		
Initial Suppression Pool pH	5.3		
Average suppression pool temperature	132°F		
Drywell Cable Data			
Hypalon Jacketed Cables	*Unit 1*	*Unit 2*	*Unit 3*
Mass of Jacket	3703 lbm	868 lbm	868 lbm
Average Cable Outside Diameter	0.89 inches	0.89 inches	0.89 inches
Average Cable Jacket Thickness	72 mils	72 mils	72 mils
Percent of Cable in Conduit	30%	50%	50%
Percent of Cable in Trays	70%	50%	50%
Polyvinyl Chloride Jacketed Cables	*Unit 1*	*Unit 2*	*Unit 3*
Mass of Jacket	NA	1389 lbm	1389 lbm
Average Cable Outside Diameter	NA	0.89 inches	0.89 inches
Average Cable Jacket Thickness	NA	72 mils	72 mils

Table 2-16 Suppression Pool pH Control Inputs			
Input/Assumption	**Value**		
Percent of Cable in Conduit	NA	30%	30%
Percent of Cable in Trays	NA	70%	70%
Neoprene Jacketed Cables	*Unit 1*	*Unit 2*	*Unit 3*
Mass of Jacket	1492 lbm	1492 lbm	1492 lbm
Average Cable Outside Diameter	0.73 inches	0.73 inches	0.73 inches
Average Cable Jacket Thickness	72 mils	72 mils	72 mils
Percent of Cable in Conduit	0%	0%	0%
Percent of Cable in Trays	50%	50%	50%
Halar Jacketed Cables[1]	*Unit 1*	*Unit 2*	*Unit 3*
Mass of Jacket	NA	155.4 lbm	NA
Average Cable Outside Diameter	NA	0.236 inches	NA
Average Cable Jacket Thickness	NA	25 mils	NA
Percent of Cable in Conduit	NA	0%	NA
Percent of Cable in Trays	NA	0%	NA
Conduit Material	Aluminum		
Conduit wall thickness	0.1 inch		
Conduit air gap	0.25 inch		

[1] Temporary cable installed in Unit 2 and planned to be removed during a future outage.

Table 2-17 Main Steam Line Break Accident Puff Release X/Q Inputs	
Input/Assumption	**Value**
Mass Release	11,975 lbm steam 42,215 lbm water (saturated @ 898psia) Assumed instantaneous release
Bubble Geometry	Spherical & Hemispherical Cases Considered
Turbine Building Perimeter Dimension	~1500 ft

The inputs for reactor building effective mixing free volume and turbine building release X/Q values differ for Unit 1 and Units 2 & 3. Accordingly, the LOCA analyses were performed separately for Unit 1 and Units 2 & 3. The impact on the Unit 1 LOCA analysis was to increase slightly the offsite doses (EAB and LPZ) because of reduced holdup in the smaller Unit 1 reactor building volume. This reduced holdup (and the use of more limiting X/Q values for the turbine building releases) would tend to also increase the control room dose for Unit 1 if activity brought into the control room was the only consideration. However, the reduced holdup (faster removal of the reactor building volume) also reduces the control room shine dose from the reactor building and the net effect is actually a small decrease in the control room dose. Because of this behavior, the reduced reactor building holdup on Unit 1 is actually a benefit to control room dose.

The results of the analyses for the Unit 1 offsite dose and the Unit 2/3 control room dose are bounding for all three units. The EAB, LPZ, and control room calculated doses are within the regulatory limits. Table 3-1 presents the results of the bounding LOCA radiological consequence analysis.

3.1.1.2 Main Steam Line Break Accident

The EAB, LPZ and control room calculated doses are within the regulatory limits for the cases analyzed. The control room doses were determined using the new X/Q value for the instantaneous puff release. The inputs used in the analyses were bounding for Units 1, 2, and 3; therefore, the results are applicable for all three units. Table 3-2 presents the results of the main steam line break accident radiological consequence analysis.

3.1.1.3 Refueling Accident

The radiological consequences of the design basis refueling accident were analyzed using a simplified configuration of one unique release pathway using the turbine building exhaust release X/Q for the EAB and LPZ, and the refueling X/Q for the control room along with the inputs/assumptions defined in Section 2.3.1.3 of this report. The inputs used in the analyses were bounding for Units 1, 2, and 3; therefore, the results are applicable for all three units. The EAB, LPZ, and control room calculated doses are within the regulatory limits. Table 3-3 presents the results of the refueling accident radiological consequence analysis.

3.1.1.4 Control Rod Drop Accident

The radiological consequences of the design basis control rod drop accident were analyzed using the RADTRAD code and the inputs/assumptions defined in Section 2.3.1.4 of this report. The inputs used in the analyses were bounding for Units 1, 2, and 3; therefore, the results are applicable for all three units. The EAB, LPZ, and control room calculated doses are within the regulatory limits. Table 3-4 presents the results of the control rod drop accident analysis.

3.1.2 Suppression Pool pH Control

The re-evolution of elemental iodine from the suppression pool is strongly dependent on suppression pool pH. The analysis assumed that sodium penteborate was injected via SLC within several hours of the onset of a LOCA. The conservative modeling of the primary containment cabling results in the production of a large amount of hydrochloric acid. The minimum suppression pool pH at 30 days post-LOCA remains above 7.0, which satisfies the conditions for inhibiting the release of the chemical form of elemental iodine in the elemental form from the suppression pool water. The suppression pool pH response over time is shown in Figure 3-1.

The quantity of SLC calculated as necessary to meet AST requirements is above the current TS requirements; therefore, TS revisions are proposed which increase the quantity of SLC required. Based on these TS changes, AST analysis for suppression pool pH control, the SLC system will be credited for limiting radiological dose following LOCAs involving fuel damage.

Table 3-1
BFN Units 1, 2, and 3
LOCA Radiological Consequence Analysis

(rem TEDE)

Dose Component	Offsite Dose[1]		Control Room Dose[2]
	EAB	LPZ	
Base of Stack	—	1.14E-2	4.49E-3
Top of Stack	—	6.14E-1	2.43E-1
Turbine Building Roof	—	3.02E-1	1.13E-1
ECCS Leakage - Base of Stack	—	1.27E-2	1.21E-2
ECCS Leakage - Top of Stack	—	3.64E-1	1.12E-1
Shine	—	N/A	7.62E-1
TOTAL	**1.11**	**1.30**	**1.25**
Regulatory Limit	25	25	5
Current Analysis (Regulatory Limit) - rem	1.67E-01 (25) Gamma 1.01E-01 (300) Beta 5.84 (300) Thyroid	4.82E-01 (25) Gamma 4.84E-01 (300) Beta 8.6 (300) Thyroid	6.83E-01 (5) Gamma 1.58E-01 (30) Beta 2.95E+01 (30) Thyroid

1 Bounding results (Unit 1) are shown.

2 Bounding results (Unit 2/3) are shown

Table 3-2 BFN Units 1, 2, and 3 Main Steam Line Break Accident Radiological Consequence Analysis (rem TEDE)			
Case	Offsite Dose		Control Room Dose
	EAB	LPZ	
3.2 μCi/gm DE I-131	1.30E-1	6.52E-2	4.09E-2
32 μCi/gm DE I-131	1.30	6.52E-1	4.09E-1
Regulatory Limit	25	25	5
Current Analysis (Regulatory Limit) - rem[1]	3.72E-01 (25) Gamma 1.56E-01 (300) Beta 2.99E+01 (300) Thyroid	1.86E-01 (25) Gamma 7.80E-02 (300) Beta 1.49E+01 (300) Thyroid	5.30E-02 (5) Gamma 3.27E-02 (30) Beta 1.05E+01 (30) Thyroid

1 Current analysis are based on 32 μCi/gm DE I-131 limit.

Table 3-3 BFN Units 1, 2, and 3 Refueling Accident Radiological Consequence Analysis (rem TEDE)			
Case	Offsite Dose		Control Room Dose
	EAB	LPZ	
24 Hours after shutdown	8.6E-01	4.3E-01	5.4E-01
Regulatory Limit	6.30	6.30	5
Current Analysis (Regulatory Limit) - rem	3.37E-01 (25) Gamma 5.77E-01 (300) Beta 3.32E+01 (300) Thyroid	1.68E-01 (25) Gamma 2.89E-01 (300) Beta 1.66E+01 (300) Thyroid	4.94E-02 (5) Gamma 4.96E-01 (30) Beta 1.74 (30) Thyroid

Table 3-4 BFN Units 1, 2, and 3 Control Rod Drop Accident Radiological Consequence Analysis (rem TEDE)			
Case	**Offsite Dose**		**Control Room Dose**
	EAB	**LPZ**	
Power Operation	1.19	6.82E-01	2.48E-01
Regulatory Limit	6.30	6.30	5
Current Analysis (Regulatory Limit) - rem	1.52 (25) Gamma 1.07 (300) Beta 1.58E+01 (300) Thyroid	8.58E-01 (25) Gamma 6.04E-01 (300) Beta 1.58E+01 (300) Thyroid	3.86E-02 (5) Gamma 4.32E-01 (30) Beta 6.3 (30) Thyroid

Table 3-5 BFN Units 1, 2, and 3 Main Steam Line Break Accident Instantaneous Ground Level Puff Release X/Q VALUE (Main Steam Line Break Accident Only)	
Time Period	Control Room (sec/m^3)
46 secs	4.60E-4

Figure 3-1: Suppression Pool pH Response