December 9, 2002

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) RELATING 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. 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 these information requests on October 15, 2002, and the requests were subsequently discussed in a teleconference on November 4, 2002.

Enclosure 1 provides TVA's response to each of the staff's questions with the exception of Request 15. As discussed with the staff in the November 4, 2002, teleconference, NRC is reconsidering Request 15.

Enclosure 2 provides replacement pages for the TS-405 submittal as referenced in the TVA responses. One regulatory commitment is contained in this response as provided in Enclosure 3.

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 this day December 6, 2002.

Sincerely,

original signed by:

T. E. Abney
Manager of Licensing
and Industry Affairs

Enclosures:

1. Response To Request For Additional Information (RAI) Relating To Technical Specifications Change No. TS-405 - Alternative Source Term (AST)
2. Technical Specifications Change No. TS-405 - Alternative Source Term (AST) - Replacement Pages
3. List of Regulatory Commitments

TVA cc: See page 3

DTL:SWA:BAB
Enclosures
cc (Enclosures):
A. S. Bhatnagar, POB 2C-BFN
M. J. Burzynski, BR 4X-C
R. G. Jones, PAB 1A-BFN
D. C. Olcsvary, LP 6A-C
C. M. Root, PAB 1G-BFN
K. W. Singer, LP 6A-C
E. J. Vigluicci, ET 11A-K
R. E. Wiggall, PEC 2A-BFN
NSRB Support, LP 5M-C
EDMS-K

s:lic/everyone/submit/subs/TS 405 AST Final RAI

ENCLOSURE 1

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

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION (RAI) RELATING TO TECHNICAL SPECIFICATIONS CHANGE NO. TS-405
ALTERNATIVE SOURCE TERM (AST)

NRC Request 1

On page 3 of the submittal letter dated July 31, 2002, TVA states that, pursuant to existing Unit 1 License Condition 2.C.(4), it will verify that the required AST analyses needed for the remaining DBAs for Unit 1 are complete and will submit them for NRC review and approval prior to the Unit 1 restart. The staff notes that the scope of the license condition applies only to Unit 1 analyses that support a technical specification change already implemented at Units 2 and 3. The license condition states that the analysis will be submitted prior to entering the mode for which the technical specification applies. The re-analysis of the three remaining DBAs is necessary to support the requested power increase and to update the UFSAR analyses to reflect the implementation of the alternative source term. It is not apparent that any particular technical specification change is involved. Please explain how the existing License Condition 2.C.(4) provides adequate assurance that the remaining three DBA analyses (i.e., LOCA, CRDA, MSLB) will be submitted for review prior to Unit 1 restart, or propose revised language for the condition that will provide this assurance.

TVA Response 1

Unit 1 was shutdown and defueled at the time of conversion to Improved Technical Specifications (ITS) in July 1998. At the time, It was recognized that extensive modifications and engineering analyses would be necessary to restart Unit 1. To ensure that appropriate controls were in place to complete activities necessary to comply with ITS for Unit 1 prior to restart, Unit 1 License Condition 2.C.(4) was established. This License Condition specifies that TVA will verify submittal of required analyses and completion of modifications needed to support the ITS as issued in Unit 1 License Amendment No. 324 and, in addition, to support any Unit 1 TS changes subsequently issued.

Accordingly, TVA considers that License Condition 2.C.(4) is likewise applicable to the TS changes being proposed in TS-405 and provides a standing obligation for TVA to submit the remaining three Design Basis Accident radiological analyses results (i.e., Loss of Coolant Accident (LOCA), Control Rod Drop Accident (CRDA), and Main Steam Line Break (MSLB)) for review prior to entering the mode for which TS-405 applies. In

TS-405, the proposed change to TS 3.1.7, Standby Liquid Control (SLC) System, adds a new requirement for SLC minimum volume that is credited in the LOCA analysis, which is applicable in Mode 3 or above. Similarly, the CRDA and MSLB events are considered possible in Modes 3 or above. Therefore, for the purposes of applying License Condition 2.C.(4), we consider that TVA has a binding commitment to provide the radiological analyses results for the three remaining accident evaluations prior to entering Mode 3 or above, which is when the revised Unit 1 TS would be in force.

If NRC is not in agreement with TVA's assessment of the requirements of the License Condition, TVA will consider a new license condition to provide any additional assurances NRC deems necessary.

NRC Request 2

On page E1-4 of Enclosure 1, TVA notes that an alternative leakage treatment path has been established using main steam system piping and the main condenser for Units 2 and 3. This discussion doesn’t address whether or not a similar approach is to be taken for Unit 1. Since this treatment path at Units 2 and 3 represent a iodine reduction greater than 90%, the conclusions drawn from the Unit 2 and 3 analyses may not support extending approval to Unit 1 without a TVA commitment to implement the necessary seismic modifications at Unit 1 and to submit the seismic ruggedness evaluation for staff review as was done for Unit 2 and 3.

TVA Response 2

Consistent with the BFN Units 2 and 3 precedent, TVA will implement seismic ruggedness modifications on Unit 1 prior to Unit 1 restart. A regulatory commitment to this effect is provided in Enclosure 3.

NRC Request 3

On page E1-5 of Enclosure 1, TVA addresses the issue of control room inleakage, by referencing a staff safety evaluation (SE) for Units 2 and 3 dated March 14, 2000. In that SER, the staff determined that there was reasonable assurance that the BFN control room would be habitable during design basis events, based on the plant configuration that existed at that time. The staff’s finding did not endorse the absolute value of the unfiltered inleakage. By the time that this amendment is implemented, three years will have past since the staff made that determination. In a parallel amendment request supported in part by the present amendment request, TVA is proposing a power uprate of about 14%. In this amendment request, TVA is proposing relaxation in testing requirements for the charcoal absorbers in the SGTS and the CREVs. These proposed amendments have the effect of reducing the margin of safety and defense-in-depth upon which the earlier staff determination was based. Therefore, please provide a statement, based on TVA’s measurements, evaluations, and actions, that supports the continued applicability of the earlier testing results as a measure of the total potential unfiltered inleakage for the control room of all three units.

TVA Response 3

TVA previously provided a detailed description of the basis for the value of unfiltered control room inleakage in a response to a RAI dated August 10, 1994 (Reference 2). By special test it was determined that the maximum unfiltered inleakage into the control building habitability zone (CBHZ) from the supply ductwork was 3717 cfm.

To be sensitive to the possibility of the unfiltered inleakage rate increasing, TVA performs follow-up inleakage surveillance testing once per 24 months. The surveillance test validates the unfiltered inleakage rate. Since the initial special test, the CBHZ has been expanded to include the possibility that a door to one of three adjoining electrical board rooms could be opened for cooling during the time that Control Room Emergency Ventilation System is relied upon. The current surveillance test includes the expanded CBHZ. Surveillance results indicate unfiltered inleakage remains bounded by the 3717 cfm value.

TVA has a program in place to control penetrations into the CBHZ and a program for maintenance of door seals. With these programs in place, TVA does not anticipate significant degradation of the CBHZ pressure boundary.

NRC Request 4

Paragraph 2.2.3 of Enclosure 4 addresses the determination of the main steam line break puff release dispersion factor. Although this section provides significant qualitative information, the staff needs additional information to completely understand and evaluate the approach, including the performance of confirmatory analyses. Table 2-17 does not appear to be sufficient. Please provide the analysis documentation, pertinent excerpts thereof, or a detailed explanation, that provides, as a minimum, the following information:

NRC Request 4.a

The significant formulae used in determining the factor and any intermediate results, and a reference or derivation for these formulae. Include formulae that were used to determine the initial conditions of the released steam puff, the bubble rise, and the transport of the puff to the control room intake.

TVA Response 4.a

The following discussion is divided into four sections: initial conditions, bubble rise, bubble transit time, and bubble dilution.

For design input data and assumptions referred to below, see the response to NRC request 4.b.

Initial Conditions

Initial conditions are identified by performing an energy balance to determine the flashing fraction,

$$mh = m_g h_g + m_l h_l$$

where m = initial liquid mass (lbm)
h = initial liquid enthalpy (Btu/lbm)
m_g = flashed steam mass (lbm)
h_g = flashed steam enthalpy (Btu/lbm)
m_l = unflashed liquid mass (lbm)
h_l = unflashed liquid enthalpy (Btu/lbm)

and the unflashed liquid and flashed steam are at atmospheric pressure and saturation temperature corresponding to atmospheric pressure (212° F).

The flashing fraction, ff, is

$$ff = m_g/m = (m_h - m_l h_l)/m/h_g$$

$$= (h - m_l h_l/m)/h_g$$

Since

$$m_l/m = (m - m_g)/m = 1\text{-}ff$$

we have

$$ff = (h - (1\text{-}ff)h_l)/h_g$$

Thus,

$$ff = (h - h_l)/(h_g - h_l)$$

Using the steam tables,

$$h(532° F) = 527 \text{ Btu/lbm}$$

$$h_l(212° F) = 180 \text{ Btu/lbm}$$

$$h_g(212° F) = 1150 \text{ Btu/lbm}$$

Thus,

$$ff = (527 - 180)/(1150 - 180)$$

$$= 0.36$$

The mass of flashed steam is m_g = 0.36 x 42,215 = 1.52E4 lbm, and the total steam mass (i.e., initial bubble mass) is 11,975 + 1.52E4 = 2.72E4 lbm.

The temperature of the mixture of released steam at 532° F and flashed steam at 212° F is

$$T_b = (1.52E4 \times 212 + 11{,}975 \times 532)/2.72E4 = 352^{\circ} F$$

$$= 812^{\circ} R$$

The initial volume of the bubble is

$$V_i = 2.72E4 \text{ lbm}/\rho_s$$

where ρ_s = 0.0307 lbm/ft^3 (steam density at 812° R)
V_i = initial volume of steam bubble (pure steam)

Thus,

$$V_i = 8.9E5 \text{ ft}^3 = 2.5E4 \text{ m}^3$$

Per Assumption 2, two bubble geometries (spherical and hemispherical) were considered to provide confidence that the result is not particularly sensitive to bubble geometry. A hemispherical bubble will have radius

$$r_h^3 = 2r_s^3$$

where r_s = radius of a sphere of equivalent volume.

Thus,

$$r_h = 1.26r_s$$

Bubble Rise

The bubble rise is caused by buoyancy. The expression for buoyancy force, due to the density difference between ambient air and the hot bubble (all steam for the no air entrainment case), is taken from Reference 3 and is

$$F_b = 144pV(1/R_aT_a - 1/R_bT_b)$$

where F_b = buoyancy force (lbf)
p = ambient pressure (psia)
V = bubble volume (ft^3)
R_a = gas constant for air
T_a = temperature of ambient air (R)
R_b = gas constant for gas in bubble
T_b = temperature of bubble (R)

Per Assumption 3, the upward velocity of the bubble is that velocity at which the drag force equals the buoyancy force. The expression for drag force is taken from Reference 4 and is

$$F_d = (1/2g_c)C_D\pi r^2 v^2 \rho_a$$

where F_d = drag force (lbf)
C_D = coefficient of drag for a sphere
r = radius of bubble (ft)
v = upward velocity of bubble (ft/s)
ρ_a = density of ambient air (lbm/ft^3)
g_c = gravitational constant (lbm-ft/lbf-sec^2)

Thus, since $F_d = F_b$,

$$v^2 = 2g_c F_b / (C_D \pi r^2 \rho_a)$$

This is the expression for bubble rise velocity.

Bubble Transit Time

The bubble transit time up to the Control Room (CR) air intake is

$$t = d / (1 \text{ m/s} * 3.28 \text{ ft/m}) \text{ seconds}$$

where d = distance from leading edge of bubble at release location to CR air intake in feet

To evaluate d, Assumptions 5 and 7 have been considered. TB failure location is assumed at any point around the perimeter of the TB. Noting that the TB perimeter is approximately 1500 feet and taking the 10% of the TB perimeter (i.e., ~150 feet) that is centered on a CR air intake (located at a corner of the TB), the distance, d, will be

$$d = 0.1 \times 1500 \times 0.5 - \text{bubble radius}$$

For the spherical bubble, this is 15.4 feet. Thus, there is 90% confidence that the distance to the CR intake is greater than 15.4 feet. As such, 15.4 feet is a reasonably conservative value for d for the spherical bubble (from Assumption 5, a bubble release location closer to the CR intake tends to reduce dilution). Thus, the transit of the leading edge of the bubble up to the CR air intake has a transit time of

$$t = d/3.28 = 4.7 \text{ seconds}$$

The transit time for the bubble across the CR air intake is calculated as

$$t = (2 \times \text{bubble radius} \times f) / 3.28$$

where f = the fraction of the bubble that has crossed the CR air intake.

The intervals referred to in Assumption 6 have been taken to be one-half of the bubble radius (i.e, 25% of the bubble diameter).

Thus, for example, for $f = 0.25$, and with spherical bubble radius of 59.6 feet,

$$t = 2 \times 59.6 \times 0.25 / 3.28 = 9.1 \text{ seconds}$$

and for $f = 0.5$,

$$t = 2 \times 59.6 \times 0.5 / 3.28 = 18.2 \text{ seconds}$$

For the hemispherical bubble, distance d is zero feet. This is because there is no pre-dilution since the leading edge of the bubble is touching the nearest CR air intake at the time of steam release. Thus, the transit of the leading edge of the bubble up to the CR air intake has a hemispherical bubble transit time of zero.

For the hemispherical bubble, the transit time for the bubble across CR air intake for $f = 0.25$,

$$t = 2 \times 75.1 \times 0.25 / 3.28 = 11.5 \text{ seconds}$$

Bubble Dilution

The plume dilution due to bubble rise is based on Assumptions 4 and 5. The swept volume as the bubble rises is the sum of the bubble volume and the volume of a cylinder with the diameter of the bubble and a height equal to bubble elevation. The dilution D (actually the inverse of dilution) is expressed as the initial (pure steam) bubble volume divided by the swept volume, i.e.,

$$D = V_i / (V_i + \pi r^2 vt)$$

where
- v = bubble rise velocity
- t = bubble rise time
- r = bubble radius
- V_i = pure steam bubble volume

The bubble volume term Vi is included in the denominator since for the case of zero rise time (same as zero transit time for the hemisphere), D should be equal to unity (i.e., no dilution).

NRC Request 4.b

A tabulation of all analysis assumptions and input values used in this assessment.

TVA Response 4.b

The design input data for the MSLB puff release dispersion factor is as follows:

Design Input Data and Assumptions

BFN Design Input Parameter	Parameter Value
Maximum time for MSIV closure	5.5 sec
Approximate volume of TB	1.5E7 ft^3
Liquid release from MSLB	42,215 lbm
Steam release from MSLB	11,975 lbm
Reference pressure for flash	898 psia (532° F)
Location of MSLB release	Into TB
Number and location of CR air intakes	2 intakes, Southwest and Southeast corners of TB
Approximate TB perimeter	1500 feet

The following assumptions were used in determination of the puff release factor for BFN:

Assumption 1 The release of steam resulting from the MSLB is instantaneous. The mass of coolant released is the amount in the steam line and connecting lines at the time of the break plus the amount passing through the MSIVs prior to closure.

Justification The BFN MSLB steam and liquid discharge is based on MSIVs closing in 5.5 seconds.

This time duration is small compared to the exposure time of interest for the CR. In any event, it is conservative to assume instantaneous release.

Assumption 2 The steam from the liquid-steam release (including the flashed steam) forms a steam bubble at ground level. The steam bubble is well-mixed. Two bubble shapes (geometries), spherical and hemispherical, are considered.

Justification The steam, due to its lower molecular weight and high temperature, will tend to displace and form a bubble in the surrounding air. As discussed below, bubble rise due to plume buoyancy is considered, but the bubble is initially assumed to be at ground level since this is conservative.

The bubble is assumed well-mixed. This is conservative since, in reality, the steam plume will be Gaussian distributed with higher activity concentrations toward the center. The rise of the bubble due to buoyancy (see Assumption 3) as it transits horizontally up to and across the CR air intake is such that the activity concentration seen by the CR air intake would be much less than the average.

Two bubble geometries are considered to take into consideration dilution sensitivity to bubble shape. Bubble geometry has a modest effect on the transit time up to and across the closest CR air intake and on the bubble rise and associated dilution. The dilution result is determined in a manner that bounds both bubble shapes.

Assumption 3 The upward velocity of the bubble is that velocity at which the drag force equals the buoyancy force.

Justification Due to the buoyancy, the bubble will rapidly accelerate upward until it reaches an equilibrium velocity at which the drag force due to friction from the surrounding air balances the buoyancy force.

Assumption 4 Bubble rise due to buoyancy is assumed to dilute the bubble contents in proportion to the volume swept by the bubble in its upward motion.

Justification The reactor coolant activity will tend to remain with the steam bubble as it rises. This would move the source term to an elevation above the CR air intake within a short time (e.g., in 20 seconds at ~15 ft/sec, the bubble will be 300 feet above the release point) such that after this time the intake would see little if any of the source term. It is, therefore, conservative to assume that the bubble contents are distributed uniformly throughout the volume swept by the bubble as it rises.

Assumption 5 The bubble elevation, and thus the dilution due to the swept volume, is based on the rise that occurs during the time required for the leading edge of the bubble to transit up to the CR air intake plus the time for the bubble to transit across the CR air intake (one diameter). Per Regulatory Guide (RG) 1.3 (Reference 5) this transit time is based on the bubble moving with a horizontal velocity of 1 meter/sec. The bubble transits across only one CR air intake (the CR air intake which is closest to the release location).

Justification As discussed further in Assumptions 6 and 7, the MSLB steam release occurs at a location slightly displaced from the CR air intake. Thus, for CR exposure to occur, the bubble must transit to the point where the leading edge of the bubble is at the CR air intake. Similarly, the CR air intake will continue to be exposed to the air with the source term until the trailing edge of the bubble passes the CR air intake.

A horizontal velocity of 1 meter/second is used based on the minimum wind speed in the Pasquill diffusion categories in RG 1.3

(Reference 5). It is noted, however, that the CR dose is determined by the integrated source term concentration (i.e., Ci–sec/m^3) which is essentially independent of this horizontal velocity (i.e., decreasing (increasing) the horizontal velocity causes a corresponding increase (decrease) in bubble rise and dilution such that the integrated source term concentration remains essentially constant).

There are two CR air intakes at the respective southeast and southwest corners of the TB. Assuming that the bubble transits across the intake closest to the release location is conservative since it minimizes the amount of bubble dilution.

Assumption 6 — The bubble dilution is taken as the average dilution at various intervals as the bubble crosses the CR air intake beginning with the dilution at the time that the leading edge of the bubble arrives at the CR air intake (pre-dilution) up to the dilution at the time that the trailing edge of the bubble leaves the CR air intake.

Justification — Depending upon the MSLB release location, the bubble starts its horizontal movement at some distance away from the CR air intake and is assumed to transit toward the CR air intake at 1 meter/second. Thus, some bubble rise, and associated dilution, which is proportional to the vertically swept volume, can occur prior to the bubble arriving at the CR air intake. This is the pre-dilution. The bubble will continue to rise, with associated dilution, as it transits across the CR air intake (a distance of one bubble diameter). The average of the dilution at various intervals as the bubble crosses the CR air intake provides the effective dilution during the period of CR intake exposure.

Assumption 7 — The primary release location (transport pathway), and that upon which the final results are based, is direct release to the environment.

Justification — Release directly to the environment is consistent with RG 1.183 (Reference 6), which states that for the MSLB accident, all the radioactivity in the released coolant should be assumed to be released to the atmosphere instantaneously.

If the TB is assumed to remain intact after the MSLB accident, the steam would be diluted by the volume (or some fraction of the volume) of the TB before beginning to leak from the TB boundary. This would be expected to give larger dilution than the direct release to the environment. Thus, it is conservative to assume direct release to the environment.

Assumption 8 — The effect of air entrainment in the bubble may be neglected.

Justification — The MSLB bubble is a transient, puff problem and the bubble will entrain increasing amounts of air as it is released and as it rises. The effect of transient air entrainment has been calculated (Reference 7) and shown to cause significant dilution. Thus, neglecting air entrainment is conservative. A non-transient calculation of the effect of air entrainment (constant amount of air entrained as bubble rises) is included in this calculation to demonstrate this effect.

Assumption 9 The steam-air mixture may be treated as a perfect gas.

Justification The perfect gas assumption is very reasonable for low pressure, high temperature gases where there are minimal interaction forces between gas molecules. This is the case for the steam-air mixture.

Assumption 10 The most conservative (maximum) CR χ/Q for the diluted steam bubble will be the dilution case (i.e., combination of bubble shape and air entrainment amount) which maximizes the product of the inverse of plume dilution and the bubble transit time.

Justification The maximum integrated activity concentration at the CR intake (Ci-sec/m^3) corresponds to the maximum χ/Q. Given a fixed amount of activity released, this integrated activity concentration is proportional to the product of plume concentration and the time that the intake is exposed to the plume. This is just equal to the product of the inverse plume dilution and the bubble transit time.

NRC Request 4.c

An explanation of the statement that no credit is taken for a vertical gradients within the rising bubble. What is meant by “leading edge of the rising bubble?” Is this a radius defined by a certain concentration percentage, e.g., 95%?

TVA Response 4.c

The steam bubble is assumed well-mixed (Assumption 2). This assumption is conservative since the actual steam plume will be Gaussian distributed with higher activity concentrations toward the center. Thus, there is a radial concentration gradient (not just vertical) in which concentration decreases significantly near the bubble edge (an activity concentration gradient). "Leading edge" refers to the bubble edge that first arrives at the CR air intake as the bubble translates from its release point (Assumption 5).

The bubble radius is not defined by a concentration percentage since the bubble activity is assumed to be well-mixed. The bubble radius is defined by the steam volume resulting from the steam directly released from the MSLB plus the amount of steam that comes from flashing of superheated liquid after release. Response 4.a provides the equations to determine initial steam volume.

NRC Request 4.d

An explanation why the puff would not be entrained in the building wake cavity. A previous turbine building dispersion factor model for BFNP assumed the release from the turbine building vents would be largely confined to the wake cavity of the turbine building.

TVA Response 4.d

Entrainment in the TB wake cavity is related to winds blowing across the TB (a distance of more than 100 meters) from the northwest to the southeast or the reverse. As noted in the discussions for NRC request 4.f, the TB can fail in any location as a result of a MSLB accident. The TB is approximately 36 meters high (above grade). Using a criterion of 2.5 times the building height as the basis for escaping the building wake, the puff reaches an elevation of 90 meters above grade between the assumed point of release from the turbine building roof and the downwind edge of the turbine building. In the BFN AST puff release analysis for an assumed spherical puff, a drag coefficient of 0.3, and no air entrainment assumed (conservative from the standpoint of the puff concentration – see TVA Response 4.b), this elevation is reached within two meters of horizontal travel for the 1.0 meter per second windspeed.

By way of examining the sensitivity of this assertion to both an increase in windspeed and a decrease in rate-of-rise, consider the following:

- If the windspeed were three times greater, this horizontal distance would increase to about five meters, but the residence time of the puff in the vicinity of the air intake would decrease correspondingly.

- To account for an increase in the aspect ratio (i.e., “flattening”) of the rising puff, (which can slow the rate-of-rise, but which will increase the swept area and the degree of dilution), the shape of the puff could either be considered to be hemispherical or the drag coefficient could be increased. Considering a hemispherical shape increases the distance necessary to reach a height of 90 meters by about 20%. Assuming a drag coefficient of unity (rather than 0.3) increases the distance necessary to reach a height of 90 meters by about 50%. In either case, the distance is less than a few meters of horizontal travel.

- Finally, one can consider that the rate-of-rise of the puff is not constant. This is because the puff (initially pure steam) will entrain air (a much colder, heavier, and denser gas), decreasing the buoyancy and slowing the rate-of-rise. It is this behavior that gives the typical height = $Kx^{2/3}$ relationship in puff rise (where x is the horizontal travel and K is a proportionality constant). In the BFN analysis, it was determined that air entrainment (by its dilution effect) actually reduces activity concentration even though it slows and eventually stops the rise of the puff. If the $x^{2/3}$ effect is included, the horizontal distance necessary to reach a 90 meter height increases by about 30%, still less than a few meters of horizontal travel.

Using a more sophisticated Polestar study model (Reference 7) for puff rise (as compared to the simplified, conservative model for puff rise used to make these calculations) results in a somewhat slower rate-of-rise than the values discussed above. However, the associated air entrainment and Gaussian activity distribution of the sophisticated model produces a substantially lower time dependent activity concentration at the location of the CR air intake than does the simplified, conservative puff model. It is important in this context to consider that the use of the simplified, conservative model results in a χ/Q for the BFN MSLB that is only a factor of four lower than that corresponding to the steam volume itself, whereas, the sophisticated model produces activity concentrations that are several orders of magnitude lower than that corresponding to the steam volume, itself.

The χ/Q for the buoyant puff may also be compared to recently promulgated NRC-sponsored models. Using the puff model from DG-1111 (as corrected by NRC in recent NRC-industry interactions on control room habitability), the BFN result would be a χ/Q of 1.4E-3 sec/m^3. This is three times greater than the submitted model χ/Q result with credit for buoyancy (4.6E-4 sec/m^3), but is 24% less than the BFN χ/Q would have been crediting only steam dilution (1.84E-3 sec/m^3). However, it is inconceivable that this steam puff, even with minor air dilution, would not rise rapidly. For a case using the submitted model with 50% air dilution (greater than the calculated dilution for the NRC model) and an assumed hemispherical geometry, the submitted model would exhibit a rate-of-rise of 22.4 meters/second. For pure steam and an assumed spherical geometry, the rate-of-rise for the submitted model would be 30.4 meters/second.

This range of initial rates-of-rise for the buoyant puff can be compared to that of another NRC (DG-1111) model, this one for buoyant plume rise. Even if the same amount of steam is assumed to be released over a 60 second period (very conservative compared to the expected 5.5 second release for the puff, but necessary to apply the plume rise model), the NRC-sponsored plume rise model would give an initial upward velocity for the plume of more than 15 meters/second. The puff rise model used in the BFN submittal (as described above) gives results for rate-of-rise as much as a factor of two greater, but that would be expected for a release of the same amount of steam over a timeframe more than ten times shorter than that assumed for the NRC's buoyant plume rise model.

The conclusion of the above discussion is that the rapid release of a steam puff from a failed TB as the result of a MSLB accident will result in a steam/air puff reaching an elevation of 90 meters above grade (2.5 times the height of the TB) within two to five meters of the location of the failure. Entrainment in the building wake in the vicinity of the CR air intakes (even for northwest or southeast winds) is, therefore, very unlikely.

If the TB were not to fail (leading to a continuous release), the χ/Q calculated using ARCON96 (as suggested by Reference 6) for the 0 – 2 hour TB ventilator release (the release would last no more than two hours) is 2.17E-4 sec/m^3 as reflected in Table 2-7 of the AST submittal. This calculated χ/Q considers building wake effects. However, the BFN puff release (χ/Q = 4.6E-4 sec/m^3) is more limiting than the continuous release result.

NRC Request 4.e

A justification of why the steam line conditions are appropriate to use for establishing the puff initial conditions, given the potential for steam energy dissipation as the steam release expands and impinges on steam tunnel and turbine building surfaces as it migrates to the assumed release point(s). The initial puff conditions are based on the mass released and temperature of the liquid/steam mixture. What is the basis for the 898 psia value for steam line condition (Submittal Table 2-17). This appears somewhat less than the turbine input pressure with the zero power end of the EHC pressure control range.

TVA Response 4.e

The potential for steam energy dissipation in the TB was considered and is discussed in Assumption 7 (see TVA response to 4.b). The primary release location (transport pathway), and that upon which the final results are based is a direct release to the environment. This is consistent with RG 1.183 (Reference 6), which states that for MSLB, all the radioactivity in the released coolant should be assumed to be released to the atmosphere instantaneously.

If the TB remains intact after a MSLB, the steam would be diluted by the volume (or some fraction of the volume) of the TB before beginning to be released from the TB ventilators. This would be expected to give larger dilution than the direct release to the environment. Thus, it is conservative to assume direct release to the environment.

Energy dissipation due to impingement of steam on TB structure and steam tunnel surfaces while the steam migrates to the release point is expected to have a small effect. This is due to two factors: 1) the short time available for energy transfer to surfaces as the bubble expands, and 2) the likelihood that partial confinement of the puff in the steam tunnel and/or TB will have a favorable effect because of dilution.

At time t = 0, the pressure for the MSLB (at hot standby conditions) is 940 psia. However, the AST analyses assume saturated water at 898 psia for the MSLB liquid release. This is the pressure at the break at the time when water starts to flow. Based on the MSLB release time histories, if the MSLB event were to occur during hot standby (which results in the enveloping radiological release) with MSIV isolation in 5.5 seconds (5 seconds for isolation and 0.5 seconds for detection), water starts to flow in approximately 1.1 seconds. These conditions maximize the water enthalpy and hence the flashing fraction. The mass release for the MSLB accident is based on choked flow through the main steam line flow restrictor. Note that these assumptions are the same as currently found for the design basis MSLB in Updated Final Safety Analysis Report (UFSAR) Section 14.6.5.

NRC Request 4.f

A justification of the 90 percent distance cutoff used in identifying release points. The discussion states that the bubble is assumed to be released at a distance from the nearest control room intake that is exceeded by 90 percent of the potential release locations. The staff notes that the resulting dispersion factor is to be used in a design basis calculation and questions why this cutoff should be considered adequately conservative.

TVA Response 4.f

The 90 percent distance cutoff is considered reasonable. As discussed in TVA Response 4.a regarding bubble transit time, it is necessary to evaluate the distance which the bubble must travel from its release point to reach the CR intake. The CR intakes are located near the southeast and southwest corners of the TB roof. A point on the TB perimeter must fail for the bubble to be released to the environment and begin its transit to the CR intake. The TB is judged equally likely to fail at any point on its perimeter (or on its roof, a possibility not considered in the determination of the 90%, but which would make the fraction of failure points excluded even smaller than 10%). It is overly conservative to assume that the TB failure point is exactly adjacent to a CR intake. Thus, the TB failure point was taken to be at a location, which is a distance of 5% of the TB perimeter (i.e, 1500 x 0.1 x 0.5 = 75 feet) away from the CR intake in either direction (i.e., 10% of the TB perimeter, which is centered on the CR intake which makes the distance ± 5%). Two additional points relative to the 10% exclusion need to be considered:

- Wind direction was not considered in the determination of the 10% probability of the χ/Q being higher than the value cited (4.6E-4 sec/m^3). The control room air intakes are located at one end of the TB at the southeast and southwest corners. Only winds from generally the northern side of a line connecting the southeast and southwest corners of the TB will transport releases from the TB in the direction of one or the other of the CR air intakes. Winds from the northern side of that line are expected only about one-half of the time. Therefore, wind direction makes the 90th percentile χ/Q actually a 95th percentile χ/Q.

- The hemispherical bubble is the controlling case for plume dilution and for the hemispherical bubble, distance “d” is zero. This is because the leading edge of the bubble is touching the nearest CR air intake at the time of steam release (due to the larger bubble radius). For this reason, the 90% cutoff has no impact on the limiting result.

NRC Request 4.g

An explanation of the bases and application of the 0.25 minimum dilution effect and why an average value is appropriate for use in the design basis analysis.

TVA Response 4.g

Based on Assumption 6 and the discussion in the response to request 4.a regarding bubble rise and bubble transit time, the effective value of “D” (puff dilution or, actually the inverse of puff dilution), is taken as the average of the bubble dilution at intervals of 0.25 times bubble diameter as the bubble crosses the CR air intake. The average dilution is calculated beginning with the dilution at the time that the leading edge of the bubble arrives at the CR air intake (no dilution for hemispherical bubble) up to the dilution at the time that the trailing edge of the bubble leaves the CR air intake.

The results are given in the table below. From Assumption 10, the most conservative CR χ/Q will result from the dilution case which maximizes the product of inverse plume dilution and bubble transit time. As is evident from the table, the hemispherical bubble with no air entrainment is the most conservative case. A plume dilution of a factor of 4 (i.e., rounded down from 4.5, which corresponds to an inverse plume dilution of 0.222) and bubble transit time of 46 seconds will be applied to the determination of χ/Q. This provides margin compared to the other cases.

Results of Plume Dilution Calculation

	Inverse Plume Dilution (a)		Bubble Transit Time [1](b)		(a) x (b)	
	Hemisph Bubble	Spher Bubble	Hemisph Bubble	Spher Bubble	Hemisph Bubble	Spher Bubble
Direct Release to Environment, no air entrainment	0.222	0.034	46	36	10	1.2
Direct Release to Environment, with bubble volume increased by 10% due to air entrainment	0.184	0.033	47	38	8.5	1.3
Direct Release to Environment, with bubble volume increased by 50% due to air entrainment	0.099	0.034	52	42	5.2	1.4

1. Bubble transient time is in seconds.

The statement that the 0.25 is an average value refers to the fact that the plume dilution starts out at unity (i.e., no dilution) with the bubble leading edge just touching the CR intake and gradually increases (due to bubble rise) as the bubble transits across the intake. To take this dilution into effect, an integration process is used. The integration process used was a coarse one as noted above (average of the bubble dilution at intervals of 0.25 times bubble diameter). The resulting χ/Q is approximately a factor of four higher than the value which would be obtained by averaging over intervals which are very small multiples of bubble diameter. However, the key here is that "average" (as used in Assumption 6) is the summation of dilution evaluated for each interval divided by the total number of intervals. This is the 0.222 value from the table above. It is not an "average" in the sense of a mean value from some distribution.

NRC Request 4.h

An explanation of how the turbine building perimeter dimension of 1500 feet (Table 2-17) factors into this assessment?

TVA Response 4.h

See TVA Response to NRC Request 4.f.

NRC Request 5

As discussed in Section 2.3.1.1 of Enclosure 4, TVA is proposing to take credit for deposition in the main steam lines and in the main condenser. Please provide the analysis documentation, pertinent excerpts thereof, or a detailed explanation, that provides, as a minimum, the following information:

NRC Request 5.a

The internal surface area of each control volume.

TVA Response 5.a

The internal surface area of the steam line control volume from the outboard MSIV to the drain line for the steam line in which the inboard MSIV is assumed to be failed open is approximately 385 ft^2. This area is identical to that for the downstream control volume of the other steam lines. The steam lines are horizontal at this location.

The internal surface area of the steam line space between the inboard and outboard MSIVs (the upstream control volume of the unfaulted line) is approximately 109 ft^2. The steam lines are also horizontal at this location.

NRC Request 5.b

An explicit numeric value for each parameter used in the assessment. The discussion refers to temperatures and pressures in non-numeric terms. This is subject to misunderstandings. For example, the text states that the pressure in the control volumes between the closed MSIVs is taken as the containment pressure. However, Table 2-12 implies that the assumption is saturated conditions at 1050 psia.

TVA Response 5.b

Under EPU LOCA conditions, the maximum drywell analyzed conditions are 48.5 psig and 295.2° F. For analysis purposes, in the volume between the inboard and outboard MSIVs, the pressure is taken as the accident pressure of the drywell (48.5 psig) and the temperature is assumed to be equal to the saturation temperature of the reactor dome at the normal operating pressure condition of 1050 psia, which is 550.6° F. These are conservatively high values as the steam line temperature is expected to drop in the main steam line as the pressure drops. In the steam lines downstream of the outboard MSIV, the pressure is assumed to be atmospheric with a temperature of 550.6° F, which continues to be a conservative representation.

NRC Request 5.c

An explanation of why the numeric values used are conservative for the entire 30 day duration of the event, e.g., as the plant cools down, and as CNMT pressure drops.

TVA Response 5.c

No credit is taken for cooldown or depressurization of the plant. The volumetric leak rates from the drywell and through the steam lines continue to be based on analyzed peak containment pressure, peak containment temperature, and maximum steam line temperature, throughout the entire 30-day duration of the event. By doing so, the activity leaked from the containment is maximized and the residence time in the steam lines is minimized.

NRC Request 5.d

An explanation of why the data derived from the staff analysis in AEB-98-03, performed for the Perry plant, are adequately representative for the BFNP configuration and operation.

TVA Response 5.d

To determine a representative settling velocity in the downstream steam line control volume for BFN (i.e., the steam line control volume not connected directly to the drywell), several conservative assumptions were made related to the distribution of settling velocities in the downstream control volume to the removal rate (i.e., effective settling velocity) in the first control volume (that connected to the drywell). The result was a substantial decrease in the settling velocity used (i.e., from the median value of 1.17E-3 m/s in the first control volume to a median value of 2.7E-4 m/s in the downstream control volume).

The NRC staff analysis of steam line removal discussed in AEB-98-03 (Reference 8) (and particularly, in Appendix A of AEB-98-03) represents steam lines between the reactor vessel and the inboard MSIV for the intact steam lines and between the MSIVs for all steam lines. The BFN analyses did not credit the portion of the steam line from the reactor vessel to the inboard MSIV for any steam line. As such, all of the piping credited for BFN is downstream of the inboard MSIVs. The space between the closed inboard and outboard MSIVs is credited for only three of the four steam lines for BFN. In the remaining steam line, the inboard MSIV is assumed to be failed open; and, therefore, this space is assumed to communicate freely with the drywell.

Because the main steam lines between the MSIVs and main condenser, and the main condenser, itself, are seismically rugged, the BFN radiological analyses also credits the portion of the steam line between the outboard MSIV and the point where the drain line flowpath to the main condenser is attached to the main steam lines. This differs from Perry. Therefore, it is recognized that the treatment of steam line deposition described in AEB-98-03 applies directly to BFN by definition only for those portions of the steam lines inside the drywell (not credited for BFN) and to the space between the MSIVs connected directly to the drywell (credited for three of the four steam lines at BFN).

Because the inboard MSIV is assumed to be failed open in the faulted steam line and it is assumed, further, that no deposition occurs in this steam line up to the outboard MSIV, the portion of this steam line between the MSIVs is considered as part of the drywell. Therefore, the portion of this faulted steam line between the outboard MSIV and the drain line tap to the main condenser is considered to be directly connected to the drywell and covered by AEB-98-03. However, in the unfaulted lines, this portion of the steam line is separated from the drywell by the portion of the steam line between the MSIVs. Therefore, AEB-98-03 is not considered to apply directly to the portion of the steam lines downstream of the outboard MSIVs for the unfaulted lines.

NRC Request 5.e

Please explain the method, assumptions, and provide the inputs that went into this assessment of the deposition velocity for the second control volume.

TVA Response 5.e

The AEB-98-03, Appendix A cumulative distribution of settling velocities represents the probability (expressed as (y+dy)-y, where y is the ordinate) that the “effective” (or mass average) settling velocity will be between x and x+dx (where x is the abscissa) in a Boiling Water Reactor (BWR) steam line under a certain set of conditions. Therefore, the greater the slope (dy/dx) at a given value of x, the greater the likelihood that the given effective settling velocity, x, would be observed.

AEB-98-03 should not be interpreted as the likelihood of finding a monodisperse aerosol with a given settling velocity. Rather, it is the likelihood that given a particular point in time and a given set of conditions, the mass removal rate of the distributed aerosol (size, density, and shape factor) will correspond to the given effective settling velocity. The conditions to which this model applies are specifically those for Perry. The applicability of those conditions to BFN is discussed in the previous section.

In Polestar’s work associated with sedimentation in a BWR steam line, it has been noted that downstream steam line control volumes (i.e., control volumes downstream from the first control volume) exhibit substantially lower sedimentation rates than steam line control volumes being fed directly from the drywell. In aerosol physics, it is usually the case that a quasi-steady airborne mean particle size is reached when the agglomeration rate of small particles (smaller than the mean) attaining the mean size becomes equal to the rate at which mean size particles are settling. Larger particles have, for the most part, already settled away. Because the sedimentation height of a BWR steam line is small, removal rates are large and the corresponding "quasi-steady" particle size is small. Therefore, in the first control volume, settling rates are high as particles larger than this quasi-steady value quickly deposit. As the particulate then moves downstream into other control volumes (assumed to be well-mixed, but at a lower concentration than the first control volume), the remaining particulate settles at a much slower rate. The corresponding effective settling velocity also decreases. Since the drywell is no longer the source of the aerosol found in these downstream control volumes, it is Polestar's view that the AEB-98-03, Appendix A conditions are no longer met and the AEB-98-03, Appendix A cumulative distribution is no longer directly applicable.

To determine a representative settling velocity in the downstream control volume several conservative assumptions to relate the distribution of settling velocities in the downstream control volume to the removal rate (i.e., effective settling velocity) in the first control volume were made. The result was a substantial decrease in the settling velocity used (i.e., from the median value of 1.17E-3 m/s in the first control volume to a median value of 2.7E-4 m/s in the downstream control volume).

NRC Request 5.f

Please explain the method, assumptions, and provide the inputs that went into the assessment of the condenser removal efficiency for particulates of 3.47E-4 m/s.

TVA Response 5.f

The same adjustments described in TVA Response 5.e were applied to the distribution of sedimentation velocities found in the upstream control volumes (main steam line volumes) for the main condenser. It was determined that the median sedimentation velocity for particulate coming from the steam line with one MSIV failed open would be 3.9E-4 m/s; and for that coming from steam lines with all MSIVs closed, it would be 0.8E-4 m/s. The reason the second value is so low is because of the very efficient removal of particulate in the steam lines with all MSIVs closed.

A weighted average of the two values (based on particulate mass leaked into the main condenser) is 3.74E-4 m/s. (Note: the 3.47E-4 m/s value listed in the NRC request above contains a transposition error).

NRC Request 5.g

Please provide a more detailed explanation of what is meant by "The steam line and main condenser removal efficiencies for particulate and elemental iodine may be combined by weighting the steam line removal according to flow and then placing these removal efficiencies in series."

TVA Response 5.g

In the RADTRAD model for BFN, all steam line leakage is combined into two flowpaths: one to the main condenser and one bypassing the main condenser. Each one of these flowpaths has elements in parallel (because of multiple steam lines) and in series (because of multiple control volumes in the steam lines with all MSIVs closed). To combine these elements, the parallel elements are combined according to a weighted average determined by the leakage flow in each element. The serial elements are combined using the expression:

$$e_{effective} = 1 - (1 - e_1)\,(1 - e_2) \bullet\bullet\bullet (1 - e_n)$$

where e is the efficiency and there are n elements in series.

NRC Request 6

In submittal Table 2-12 of Enclosure 4, there appears to be an error in the early in-vessel fraction for cerium and lanthanide. The total fraction is correct. Please resolve.

TVA Response 6

There is a typographical error in the in-vessel fraction for cerium and lanthanides in submittal Table 2-12. The correct numbers should be 0.0005 for the cerium group and 0.0002 for the lanthanides. The correct values were used in the BFN AST analyses. A corrected replacement Table 2-12 is provided in Enclosure 2.

NRC Request 7

In submittal Table 2-12 of Enclosure 4 tabulates the reactor building *free* volume as 1,931,502 ft^3 with a statement "50% of this value is used due to incomplete mixing." BFNP UFSAR (BFN-19) Section 14.6.3.6 states that the *effective* mixing volume *is* 1,931,502 ft^3. The proposed UFSAR changes add a note that states that the value represents 50% of the total secondary containment volume. An earlier (BFN 15) version of the UFSAR stated that the effective volume of the secondary CNMT is 50% of the total free volume of a single reactor zone and 50% of the refueling zone, *resulting in 1,931,502 ft^3*. This explanation was also given in an RAI response for a prior amendment. The staff believes that the 1,931,502 ft^3 value given in submittal Table 2-12 already includes the 50% adjustment called for in the table. Please resolve and confirm that the correct value was used in the calculations.

TVA Response 7

The correct value of 1,931,502 ft^3 was utilized in the LOCA calculation. This volume represents 50% of the free volume of the refueling floor plus 50% of the free volume of the Unit 2 or 3 reactor building. Therefore, the 1,931,502 ft^3 value given in submittal

Table 2-12 does already includes a 50% adjustment for incomplete mixing. A revised replacement Table 2-12 and UFSAR page 8 is provided in Enclosure 2 to clear up this point.

NRC Request 8

Table 2-14 of Enclosure 4 states that the elemental pool DF is 500 and the organic DF is 1.0. In Appendix B of RG 1.183, the staff stated that the decontamination factors for the elemental and organic species are 500 and 1, respectively, giving an overall effective decontamination factor of 200. It was the staff's intent that the effective DF be limited to 200. The use of the individual DFs will yield an effective DF of about 280. The staff erred in including the individual DFs in Appendix B and these will be removed in a future revision. If TVA used the factors individually, the staff requests that TVA make a commitment that the BFNP Licensing Basis will incorporate the effective DF of 200 and that future re-analyses will use that effective value. The staff believes, given the fuel handling accident doses tabulated in Table 3-3, that the revised doses will still be less than the acceptance criteria. We will confirm this during our review.

TVA Response 8

The dose calculations provided in the July 31, 2002 (Reference 9), TS-405 submittal were based on decontamination factors (DF) of 500 and 1 as prescribed in RG 1.183 Appendix B (Reference 6). In the telecon on November 4, 2002, TVA indicated that the more conservative DF (200) intended by RG 1.183 would be utilized the next time the subject dose calculation were reperformed.

However, for completeness and to be responsive to NRC's request, TVA has elected to revise the calculation for the FHA radiological dose to use the overall effective DF of 200. The revised control room and offsite doses increase slightly, but all doses remain less than 1 rem Total Effective Dose Equivalent. Replacement pages for the TS-405 submittal to reflect this calculation revision are provided in Enclosure 2.

NRC Request 9

Table 2-14 of Enclosure 4 tabulates the "maximum" drywell accident conditions as being 48.3 psig and 294.9 degrees F. UFSAR §14.6.3.3 identifies the primary containment requirements as 56 psig and 281 degrees F. Please resolve the differences and confirm that the appropriate value was used in the calculations.

TVA Response 9

The design parameters for the primary containment are 56 psig and 281° F.

The values used in the AST analyses are the peak calculated drywell accident analyses values (48.3 psig and 294.9° F). The EPU containment accident analysis final values were later determined to be 48.5 psig (peak drywell pressure) and 295.2° F (peak drywell gas temperature). The peak drywell gas temperature exceeds the drywell shell design temperature for a short period at the beginning of the accident. This is not considered a threat to the drywell shell structure due to the short duration of the increase relative to the time required for the drywell shell heatup. An evaluation of the AST analyses determined that the final radiological dose consequences were not affected by this small difference in EPU drywell conditions. Table 2-12 provided in Enclosure 2 has been updated to reflect these values.

NRC Request 10

Table 2-15 of Enclosure 4 provides a release period of 24 hours. However, Figure 2-2 shows a 30 day release period. Please resolve this apparent inconsistency.

TVA Response 10

This 24-hour value in Table 2-15 is incorrect and has been revised to indicate a release period value to 30 days consistent with Figure 2.2. A revised Table 2-15 is provided in Enclosure 2

NRC Request 11

Figure 2-1 shows a path (1) for drywell/torus mixing after release. The parameterization of this path was not addressed in the text or in the input tables. Please provide the flow rates or transfer rates used for this path and their basis.

TVA Response 11

Flow from the drywell to the torus is ignored during the initial phase of the LOCA (core damage activity release phase from time zero to approximately two hours). An assumption of no flow during this time period is conservative since activity available for drywell leakage outside containment via the MSIVs is maximized. At two hours, when core quenching is assumed to occur, there will be substantial steam production in the reactor vessel and drywell that will purge a large fraction of the drywell atmosphere though the torus downcomer vents, through the suppression pool water, and into the torus air space. If the purged drywell activity were then assumed to remain in the torus, activity available for drywell leakage outside containment via the MSIVs would be correspondingly reduced. Thus, even though the sequestering of most of the purged drywell activity in the torus would actually be expected, a well-mixed torus air space and drywell is conservatively assumed in the analysis. Therefore, after the end of the quenching period and core damage activity release phase (t = 2.033 hours), for analysis purposes, the drywell/torus mixing flowrate path is set high (assumed flow rate of one torus air volume per minute) to model a uniform distribution of activity in the drywell and torus air space. As noted above, the assumption of a well-mixed containment (drywell and torus air space) after the quenching of the core and the drywell purge is complete is conservative.

NRC Request 12

Figure 2-2 shows the mechanical vacuum pump release path for the CRDA. The previous BFN analysis assumed a release path direct from the main condenser and one via the SJAE and offgas system (UFSAR 14.6.2.7) Please explain why these two paths are no longer addressed.

TVA Response 12

The CRDA was evaluated for AST for the same cases and release points as discussed in the UFSAR 14.6.2.7. The mechanical vacuum pump (MVP) release path case continues to be the worst case under AST. Therefore, only the results of the MVP release path were provided in were reported in the July 31, 2002, TS-405 AST submittal Safety Assessment.

NRC Request 13

Section 2.3.2 of Enclosure 4 addresses the use of SLC for pH control. The discussion on page 16 states that the operator will initiate SLC based upon an alarm response procedure. The staff notes that there are steps in the generic BWROG ERGs that direct the operator to terminate boron injection. For example, in the tree for RC/Q, the continuing action block contains a direction: "If while executing these steps: It has been determined that the reactor will remain shutdown under all conditions without boron, terminate boron injection and enter [scram procedure]." Similarly, SAGs RC/F-2, RC/F-3 also have steps calling for termination of all injection.

NRC Request 13a

Please explain how the instructions in the alarm response procedure will ensure that the injection for pH control occurs when needed when the ERGs/SAGs appear to direct otherwise.

TVA Response 13a

AST implementation involves changing the appropriate BFN Alarm Response Procedure to require Standby Liquid Control System (SLCS) injection on indication of high drywell radiation. This procedure change will also necessitate changes to the Emergency Operating Instructions (EOI). Specifically, the EOI Reactor Power Control (RC/Q) tree will be revised such that SLC injection will not be terminated if it is required by the BFN Alarm Response Procedure for high drywell radiation.

BFN severe accident management guidelines (SAMGs) are consistent with industry guidelines. BFN SAMG-1 strategies PC/F-2 through PC/F-6 require termination of injection into the RPV from sources external to primary containment if the primary containment pressure limit is reached, <u>except SLC boron injection</u>. Thus, the SAMGs require continued SLC injection and do not need to be revised for AST implementation.

NRC Request 13b

Discuss other containment Radiation Monitoring indications the operator would have for determining the need for SLC initiation. Discuss redundancy of the indication.

TVA Response 13b

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 13c

Section 2.3.2 of Enclosure 4 addresses the use of SLC for pH control. You described the SLC system as a Special Safety system. Explain the difference between Special safety and Safety related.

TVA Response 13c

Special safety system are designed to respond to special events. Special events are those postulated to demonstrate some special capability of the plant or its systems (see definition 17 UFSAR Section 1.2). Special events are low probability events that are not considered accidents or abnormal events. For BFN inability to shut down with control rods, Anticipated Transient Without Scram (ATWS), control room abandonment and Appendix R fire events are considered special events. Equipment credited with responding to design basis transients and accidents have a prescribed set of general requirements. Special systems are designed, constructed and maintained with a limited set of those requirements consistent the importance of the function.

The Standby Liquid Control System (SLCS) is classified as a special safety system because it is currently designed to respond to shutdown without control rods and ATWS. FSAR section 3.8 describes the qualification required for the SLCS. Some of the differences between these qualifications and those required for a safety related system are:

- No physical separation

- Environmental Qualification is not maintained in accordance with the maintenance rule program

- Limited Quality Assurance is applied

- Strict single failure/redundancy (pumps, power supplies and valves are redundant)

Using the SLCS to control pH in the suppression pool following a postulated LOCA with fuel damage constitutes a new SCLS 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
- Governed by Technical Specifications, Limiting Conditions of Operation, and subject to Surveillance Requirements
- Simple equipment design
- No components inside containment other than piping and check/manual valves
- Quality Assurance attributes that provide hi level of assurance.
- Diverse system status indication for pumps and other components
- Redundant components receive electrical power from electrical buses that are connected to standby power supply system (AC or DC as applicable)
- Designed to maintain Reactor Coolant Pressure Boundary and Primary Containment Boundary
- Subject to American Society of Mechanical Engineers Section XI In Service Inspection requirements

NRC Request 14

In the submittal the licensee described a method used for controlling pH of the suppression pool. The method consists of using buffering action of the sodium pentaborate from the Standby Liquid Control System. The licensee determined the amount of the sodium pentaborate required to produce sufficient buffering action to counteract the effect of acidic species produced in radiation environment. In order the review the licensee methodology the staff require detailed description of the licensee's analysis. The description should include:

- generation of hydrochloric acid by decomposition of the chlorine bearing cables
- production of nitric acid in the post accident radiation field
- determination of the amount of sodium pentaborate required for maintaining the suppression pH below 7. (the licensee should provide the input and output to the computer code used in the analysis).

TVA Response 14

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 Hypalon jacket = 868 lbm
9. Thickness of Hypalon jacket = 0.072 inch
10. Mass of PVC jacket = 2865 lbm

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

11. Thickness of PVC jacket = 0.072 inch
12. Cable OD = 0.89 inch
13. Fraction of cable in conduit = 30 %
14. Air gap in conduit = 0.25 inch, see Assumption below
15. Conduit wall thickness = 0.1 inch, see Assumption below
16. Conduit material = aluminum, see Assumption below
17. Drywell free volume = 159,000 ft^3
18. Minimum torus free volume = 119,400 ft^3
19. Volume of sodium pentaborate in SLCS = 4000 gal.
20. Sodium pentaborate concentration in SLCS = 8 weight %
21. Density of SLCS containing 8 weight % sodium pentaborate = 8.64 lbm/gal.
22. Chemical formula for sodium pentaborate = $Na_2O \bullet 5B_2O_3 \bullet 10H_2O$
23. Boron enrichment in sodium pentaborate is 62.9 mole % B^{10}
24. Drywell coating = 28,780 ft^2 epoxy coating
25. 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.

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

Time	$[HNO_3]$	Net $[OH^-]$	[HCl]	$[H^+]$ Added	Net $[H^+]$ Added
1h	5.87E-6	1.19E-4	1.32E-5	1.91E-5	(1.06E-4)
2h	8.06E-6	1.17E-4	2.49E-5	3.30E-5	(9.21E-5)
5h	1.26E-5	1.12E-4	5.28E-5	6.54E-5	(4.66E-5)
12h	2.00E-5	1.05E-4	9.92E-5	1.19E-4	(5.80E-6)
1d	2.98E-5	9.49E-5	1.58E-4	1.88E-4	6.31E-5
3d	5.75E-5	6.71E-5	3.17E-4	3.75E-4	2.50E-4
10d	1.09E-4	1.57E-5	5.49E-4	6.58E-4	5.33E-4
20d	1.42E-4	(1.72E-5)	6.39E-4	7.81E-4	6.56E-4
30d	1.63E-4	(3.86E-5)	6.69E-4	8.32E-4	7.08E-4

Note: Data in parentheses indicate a negative value in the ion balance

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. The "Net $[H^+]$ Added" is the difference between the HCl and the "Net $[OH^-]$". Therefore, if "Net $[H^+]$" is used to calculate pH, then the favorable effects of CsOH are considered. If "$[H^+]$ Added" is used to calculate the pH, then the favorable effects of CsOH are ignored. "$[H^+]$ Added" is used to calculate the pH for BFN. However, 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

<u>NRC Request 15</u>

The full implementation of AST analyses will modify the licensing bases by adopting AST methodology which replaces the current accident source term with an alternative source term as prescribed in 10 CFR 50.67 and establishes the 10 CFR 50.67 total effective dose equivalent (TEDE) dose limits as a new acceptance criteria. Provide a discussion of the impact on environmental qualification (EQ) based on the doses using the AST. In addition, discuss the impact of the postulated increase in the cesium concentration 30 days following the an accident with regard to the calculated dose and the component's qualification dose.

<u>TVA Response 15</u>

TVA will provide a response to this request at a later date contingent on additional clarification from NRC.

<u>NRC Request 16</u>

Please assess the seismic capabilities of structures, systems, and components (SSC) related to the AST where credit has been taken for the flow-path of offsite release. Please provide the detailed results that ensure the integrity and function of SSC for the safe-shutdown earthquake.

TVA Response 16

The primary containment system is designed as a Seismic Class I system (UFSAR Section 5.2.3.1). The Reactor Building and the Standby Gas Treatment Building, including the plant stack, are Seismic Class I Structures (UFSAR Section 12.2.4). The Standby Gas Treatment System and Secondary Containment System are designed as Seismic Class I systems except for the penetrations through the secondary containment membrane. These penetrations are designed to limit the inleakage flow in order to maintain a negative pressure inside secondary containment following a Design Basis Earthquake (UFSAR Section 5.3.2).

The Standby Gas Treatment system is the SSC credited for the offsite flow path for effluent from the primary and secondary containment following a LOCA. The MSIV installation is designed as Seismic Class 1 (UFSAR Section 4.6.3).

The main steam lines from the outboard MSIVs and the main condenser are seismically rugged on Units 2 and 3 and are designed to remain intact following a safe-shutdown earthquake. (Enclosure 3 provides a commitment that the corresponding Unit 1 equipment will be made seismically rugged prior to Unit 1 restart). MSIV leakage from the primary containment passes through the steam lines into the main condenser, passes through the main condenser before release through the turbine building roof.

By application dated September 28, 1999 (Reference 12), as supplemented February 4, 2000 (Reference 13), TVA requested a revision to the TS to increase the allowable leakage for the main steam lines on Units 2 and 3. This license amendment made use of a previously approved General Electric methodology for providing an alternative leakage path from the MSIVs to the main condenser. An alternate leakage flow path to the main condenser using the main steam drain piping and the main condenser was established and credited using NEDC-38858 (Reference 11). By letter dated March 14, 2000, NRC approved the TS change request (Reference 1). The seismic ruggedness aspects of this alternate flow path are addressed in detail in the referenced submittals and are credited in the current UFSAR LOCA analysis regarding dose analyses for releases. The AST analysis provided in TS-405 did not credit any seismic features beyond those currently credited in the UFSAR accident analyses.

References:

1. NRC Letter to TVA dated March 14, 2000, "Browns Ferry Nuclear Plant (BFN) - Issuance of Amendments regarding Limits on Main Steam isolation Valve Leakage (TAC NOS. MA6405 and MA6406)"

2. TVA letter to NRC dated August 10, 1994, "Browns Ferry Nuclear Plant (BFN) - Response to Additional Information Regarding the Control Room Emergency Ventilation System (CREVS)" (Tac NOS. M83348, M83349 and M83350)

3. Marks Standard Handbook for Mechanical Engineers, Ninth Ed., McGraw-Hill, page 3-41

4. R. Sabersky et al, Fluid Flow, 2nd ed., Macmillan Company, pages 168-171

5. Regulatory Guide 1.3, “Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors,” U.S. NRC, Rev. 2, June, 1974

6. Regulatory Guide 1.183, “Alternative Radiological Source Terms For Evaluating Design Basis Accidents at Nuclear Power Reactors”, USNRC, July 2000

7. Haihua Zhao, “Non-Safety Calculation of Iodine Concentration at a Receptor from an MSLB Accident Using a Single Puff Model,” prepared for Polestar Applied Technology, Inc., Polestar proprietary document, July 2001

8. AEB-98-03, “Assessment of Radiological Consequences For Perry Pilot Plant Application Using the Revised (NUREG-1465) Source Term,” dated December 9, 1998.

9. TVA Letter to NRC dated July 31, 2002, "Browns Ferry Nuclear Plant - Units 1, 2, and 3 - License Amendment - Alternative Source Term"

10. PSAT C107.02, STARpH, A code for Evaluating Containment Water Pool pH During Accidents, Code Description and Validation and Verification Report, Revision 4, February 2000.

11. General Electric NEDC-38858, "Boiling Water Reactor Owners Group Report for Increasing Main Steam Isolation Valve Leakage Rate and Elimination of Leakage Control Systems," Dated October 1991

12. TVA Letter to NRC dated September 28, 1999, "Browns Ferry Nuclear Plant (BFN) - Units 2 and 3 - Technical Specifications (TS) Change 399 - Increased Main Steam Isolation Valve (MSIV) Leakage Rate Limits and Exemption From 10 CFR 50 Appendix J"

13. TVA Letter to NRC dated February 4, 2000, "Browns Ferry Nuclear Plant (BFN) - Units 2 and 3 - Response to Additional Information Regarding Technical Specifications (TS) Change No. 399 - Increased Main Steam isolation Valve (MSIV) Leakage Rate Limits and Exemption From 10 CFR 50 Appendix J - Revised TS pages For Increased MSIV Leakage Limits"

14. Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," Rev 3, Dated May 1983.

ENCLOSURE 2

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

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION (RAI) RELATING TO TECHNICAL SPECIFICATIONS CHANGE NO. TS-405
ALTERNATIVE SOURCE TERM (AST)

REPLACEMENT PAGES

This Enclosure provides replacement pages for the TS-405 Submittal made on July 31, 2002. A line has been drawn in the right margin indicating a change.

Refueling Accident

For the AST design basis refueling accident the EAB, LPZ, and control room calculated doses are within the regulatory limits. The results are summarized in the table below along with the results of the current source term analyses.

Refueling Accident Radiological Consequence Analysis (rem TEDE)			
	Offsite Dose		Control Room Dose
Case	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 2-12
LOCA Inputs

Input/Assumption	Value
Fission Products Release Fractions	Regulatory Guide 1.183 Table 1 **BWR Core Inventory Fraction Released Into Containment** **Group** / **Gap Release Phase** / **Early In-vessel Phase** / **Total** Noble Gases 0.05 0.95 1.0 Halogens 0.05 0.25 0.3 Alkali Metals 0.05 0.20 0.25 Tellurium Metals 0.00 0.05 0.05 Ba, Sr 0.00 0.02 0.02 Noble Metals 0.00 0.0025 0.0025 Cerium Group 0.00 0.0005 0.0005 Lanthanides 0.00 0.0002 0.0002
Fission Product Release Timing	Regulatory Guide 1.183 Table 4 **LOCA Release Phases BWR** Phase / Onset / Duration Gap release 2 min 0.5 hr Early In-Vessel 0.5 hr 1.5 hr
Fission Product Iodine Chemical Form	Particulate 95% Elemental 4.85% Organic 0.15%
Control Room Isolation/CREV Initiation	10 minutes
ECCS Leakage Release Fractions	Ten percent of the radioiodine in the leaked coolant is assumed to become airborne in the reactor building (secondary containment). Of this activity, 97% is assumed to be elemental iodine and 3% is assumed to be organic iodine.
Flow Rates	
Primary Containment Leak Rate (30 days)	2 % containment air weight/day
Secondary Containment Bypass Leak Rate (30 Days)	HWWV = 10 scfh beginning at t>8 hours
Assumed ECCS Leak Rate (30 days)	5 gpm
ECCS Leakage Temperature	<212°F

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

Table 2-13 Main Steam Line Break Accident Inputs	
Input/Assumption	**Value**
Mass Release	11,975 lbm steam 42,215 lbm water (saturated @ 898psia)
MSIV Isolation Time	5.5 seconds
DE I-131 Equilibrium Value	3.2 µCi/gm
DE-I-131 Pre-Accident Spike	32 µCi/gm (Conservative to TS value of 26µCi/gm)
Iodine Species Release Fraction	All Assumed Elemental

Table 2-14 Refueling Accident Inputs	
Input/Assumption	**Value**
Number of Failed Rods	111
Radial Peaking Factor	1.5
Fuel Decay Period	24 hours
Overall Effective Decontamination Factor for Organic and Elemental Iodine	200
Release Period	Instantaneous
Reactor Building Ground Release Location	Reactor Building Refueling Zone Vent (No credit for holdup or SGT operation)
Release Fractions	Noble Gases excluding Kr-85 5 percent Kr-85 10 percent I-131 8 percent Iodines except I-131 5 percent

Table 2-15
Control Rod Drop Accident Inputs

Input/Assumption	Value
Number of Failed Rods	850
Percent Fuel Melt for Failed Rods	0.77 %
Radial Peaking Factor	1.50
Release Period	30 Days
Main Condenser and Low Pressure Turbine Free Volume	187,000 ft^3
Stack Room Volume	69,120 ft^3 (50% of this value used due to incomplete mixing)
Assumed Base of Stack Leakage	10 cfm
Mechanical Vacuum Pump Flowrate	1850 scfm @ 7" Hg
Gap Release Fractions	Noble Gas 10% Iodine 10% Br 5% Cs, Rb 12% Te Group 0% Ba, Sr 0% Noble Mtls 0% Ce Group 0% La Group 0%
Core Melt Release Fractions	Noble Gas 100% Iodine 50% Br 30% Cs, Rb 25% Te Group 5% Ba, Sr 2% Noble Mtls 0.25% Ce Group 0.05% La Group 0.02%
Activity that reaches 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 2-15 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
Mass of Polyvinyl Chloride Jacket in the Drywell	2865 lbm
Mass of Hypalon Jacket in the Drywell	868 lbm
Average Cable Outside Diameter	0.89 inches
Average Cable Jacket Thickness	72 mils
Percent of Drywell Cable in Conduit	30%
Conduit Material	Aluminum
Conduit wall thickness	0.1 inch
Conduit air gap	0.25 inch

Table 3-2 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

Table 3-3 Refueling Accident Radiological Consequence Analysis (rem TEDE)			
Case	Offsite Dose		Control Room Dose
	EAB	LPZ	
24 Hours after shutdown	8.6-01	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

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

ventilation system is turned off and the Standby Gas Treatment System (SGTS) is initiated as a result of low reactor water level, high drywell pressure, or high radiation in the Reactor Building. Any fission product removal effects in the secondary containment such as plateout are neglected. The fission product activity released to the environs is dependent upon the fission product inventory airborne in the secondary containment, the volumetric flow from the secondary containment, and the efficiency of the various components of the SGTS.

The following assumptions were used to calculate the fission product activity released to the environment from the secondary containment:

a. The primary containment atmosphere leakage to secondary containment mixes instantaneously and uniformly within the secondary containment.

b. The effective mixing volume of the secondary containment is 1,931,502 ft^3 (50% of the free volume of the refueling floor plus 50% of the free volume of the Unit 2 of 3 reactor building. This volume includes a 50% adjustment for incomplete mixing

c. The SGTS removes fission products from secondary containment. If only two of the SGTS trains are in operation (i.e., SGTS flow of 16,200 cfm), a short period exists at the start of the accident during which the secondary containment becomes pressurized relative to the outside environment. ~~During this short time period, a very small amount of secondary containment atmosphere (~35 ft³) will be released directly to the environment unfiltered from the Reactor Building.~~ However, negative pressure would be re-established in secondary containment prior to fission product release times specified by RG 1.183. Once the secondary containment pressure is reduced below atmospheric pressure, all releases from secondary containment to the environment are through the SGTS filters via the plant stack. If all three trains of SGTS are in operation (i.e., SGTS flow of 24,750 cfm), all releases to the environment from secondary containment are through the SGTS filters via the plant stack. The case with three trains in operation is the limiting condition.

d. The Containment Atmospheric Dilution (CAD) System operates for a period of 24 hours at a flow rate of 139 cfm at 10 days, 20 days, and 29 days post accident. This flow is filtered via the SGTS filters.

e. The ECCS systems leak reactor coolant directly to the secondary containment. The maximum water temperature is ~~177~~ less than 212°F. The ~~ECCS~~ volume available for mixing is ~~141,260~~ 1.41E5 ft^3. Ten percent of the iodine in the ECCS ~~water~~ leakage is assumed to become airborne.

f. Filter efficiency for the SGTS was taken as 90 percent for organic and 0% inorganic (elemental) iodine.

g. Release to the environment from the plant stack is composed of three flow paths. A continuous ground level release of 10 cfm occurs at the base of the stack. This flow results from SGTS leakage through the backdraft dampers in the base of the stack. Subsection 5.3.3, "Secondary Containment System

b. Iodine Decontamination Factor in Reactor Cavity Pool Water — ~~100~~ 200 elemental and organic

c. Iodine Species: — 99.85% elemental, 0.15% organic

14.6.4.5 Fission Product Release to Environs

The following assumptions and initial conditions are used in calculating the dose existing at the exclusion area boundary ~~and~~ , at the low population zone, and to the control room operators due to fission product release.

~~a. High radiation levels in the reactor building will isolate the normal ventilation system and actuate the Standby Gas Treatment System. The isolation dampers were assumed to close in 10 seconds.~~

~~b. Since the refueling accident does not result in the release of any liquid or vapor to the secondary containment, the normal environmental condition existing prior to the accident will also exist after the accident except for the addition of the released fission products. The relative humidity in the secondary containment will, therefore, be considerably below any levels which may be detrimental to the Standby Gas Treatment System. However, air flowing through the filter system, is heated approximately 14°F above the mixture entering the system, reducing the relative humidity from 100 percent to 70 percent or less.~~

~~c. Standby Gas Treatment System Filter Efficiency — 0.90~~

~~d. Height of the Main Stack — 183 meters~~

~~e. Distance to Exclusion Area Boundary — 1,465 meters~~

~~f. Distance to Low Population Zone — 3,200 meters~~

~~g. Mixing Air Volume — 4,900 FT^3~~

~~h. Ventilation Air Flow Prior to Damper Isolation — 20,000 CFM~~

a. The release is assumed to be an instantaneous ground level release to the environment with no holdup time in secondary containment. Accordingly, no credit is taken for filtering by the standby gas treatment system and no credit is taken for an elevated release at the main stack.

b. No credit is taken for isolation of the control room nor for any filtering by the control room emergency ventilation system.

c. The X/Q for the control room is reduced by 50% to reflect the credit for dual control room air intakes as allowed by Standard Review Plan Section 6.4.

d. Control Room Free Volume 210,000 ft^3

The design basis fuel handling accident assumes that during the refueling period a fuel bundle is dropped into the reactor cavity pool. The dropped fuel bundle strikes additional bundles in the reactor core fracturing 111 fuel pins (assuming GE 7x7 fuel design). ~~Ten percent of the halogen isotopes inventory plus 10 percent of all noble gases inventory (except Kr 85 which is 30 percent of this inventory)~~ The inventory described above will be released from the fractured fuel rods. An overall decontamination factor of ~~100~~ 200 is applicable for iodine released at depth under water. The radioactive releases to the air space above the pool are released ~~through the refueling zone ventilation and the Standby Gas Treatment Systems~~ instantaneously to the atmosphere with no holdup in secondary containment and no filtering by the standby gas treatment system. The assumptions used to evaluate the fuel handling design basis accident event are defined in Nuclear Regulatory Commissions Regulatory Guide ~~1.25~~ 1.183. Further guidance is contained in the standard review plans in NUREG-800, Section ~~15.7.4~~ 15.0.1.

~~In order to evaluate the effect of refueling zone ventilation damper closure time, the analysis includes doses from air bypassing the Standby Gas Treatment System. The bypass is occurring through the Refueling Zone Ventilation System. For this evaluation, it is assumed that the portion of the ventilation system dedicated to the reactor vessel pool and the spent fuel storage pool provides the bypass flow. The gases released from the damaged fuel bundles are assumed to be confined to an air volume bounded by the perimeter of the pool and mixed to a height of no more than 4 feet above the pool. The activity released to the environment before the dampers close is taken from the air volume over the pool expelled through the ventilation system.~~ The total activity released is greater for a fuel handling accident in the reactor cavity pool than for an accident in the fuel storage pool. Normally, the number of fuel rods fractured in a drop into the reactor vessel pool is slightly larger than the number of rods fractured in a drop into the storage pool. This provides a bigger source for the vessel event. ~~However, the ventilation flow from the storage pool area is twice the size of the flow from the reactor vessel area. The difference in flows transports more activity to the environment in a given time period. Therefore, for conservatism the number of rods damaged and resulting activity released is based on a fuel handling accident in the reactor cavity, and the mixing volume and ventilation is based on a release over the spent fuel pool.~~

~~The bypass flow not only bypasses the SGTS filters, it is also released from a roof vent rather than the main stack. The atmospheric dispersion, X/Q, of releases from the top of the stack is significantly smaller than the atmospheric dispersion factors for the roof vent releases. The result of this change is to make the dose contribution from the roof~~

ENCLOSURE 3

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

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION (RAI) RELATING TO TECHNICAL SPECIFICATIONS CHANGE NO. TS-405
ALTERNATIVE SOURCE TERM (AST)

LIST OF REGULATORY COMMITMENTS

1. TVA will complete seismic ruggedness analyses and modifications on Unit 1 prior to Unit 1 restart.