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ACCESSION NBR:9806220196 DOC.DATE: 98/06/12 NOTARIZED: NO DOCKET #
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50-296 Browns Ferry Nuclear Power Station, Unit 3, Tennessee 05000296
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SUBJECT: Provides response to request for addl info re Units 2 & 3 TS Change TS-384 for power update operation.

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June 12, 1998

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

Gentlemen:

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

BROWNS FERRY NUCLEAR PLANT (BFN) - RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION (RAI) RELATING TO UNITS 2 AND 3 TECHNICAL SPECIFICATION (TS) CHANGE NO. TS-384 - POWER UPRATE OPERATION (TAC NOS. M99711 AND M99712)

This letter provides additional information requested by NRC in support of TS-384. On October 1, 1997, TVA provided TS-384, an amendment to Operating Licenses DPR-52 and DPR-68 that will allow Units 2 and 3 to operate at an uprated power level of 3458 MWt.

Enclosure 1 provides TVA's response to the May 7, 1998, NRC RAI for the October 1, 1997, proposed TS change. This letter includes replies to each of the NRC requests. Additionally, revised pages of the October 1, 1997 submittal as discussed in Enclosure 1 are included as Enclosure 2.

D030

Based on the telephone conversation with your staff on June 1, 1998 regarding the control room emergency ventilation system related to the power uprate license amendment, TVA proposes the following operating license condition:

> TVA will perform an analysis of the design basis loss of coolant accident to confirm compliance with General Design Criteria-19 and offsite dose limits considering main steam isolation valve leakage and emergency core cooling system leakage. The results of this analysis will be submitted to the NRC for

review and approval by March 31, 1999. Following NRC approval any required modifications will be implemented during the refueling outages scheduled for Spring 2000 for Unit 3 and Spring 2001 for Unit 2.

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

Sincerely,

T. E. Abney
Manager of Licensing
and Industry Affairs

cc: See Page 4

REFERENCES

1. TVA letter to NRC dated October 1, 1997, Browns Ferry Nuclear Plant (BFN) - Units 2 and 3 - Technical Specification (TS) Change TS-384 - Request For License Amendment for Power Uprate Operation

2. TVA letter to NRC dated March 16, 1998, Browns Ferry Nuclear Plant (BFN) - Units 2 and 3 Technical Specification (TS) No. 384 Supplement 1 - Request for License Amendment for Power Uprate Operation

3. NRC letter to TVA dated May 7, 1998, Browns Ferry Plant Units 2, and 3: Request for Additional Information Relating to Technical Specification Change No. TS-384 - Power Uprate Operation (TAC Nos. M99711 and M99712)

Enclosures
cc (Enclosures):

Albert W. De Agazio, Project Manager
U.S. Nuclear Regulatory Commission
One White Flint, North
11555 Rockville Pike
Rockville, Maryland 20852

Mr. Harold O. Christensen, Branch Chief
U.S. Nuclear Regulatory Commission
Region II
101 Marietta Street, NW, Suite 2900
Atlanta, Georgia 30323

NRC Resident Inspector
BFN Nuclear Plant
10833 Shaw Road
Athens, Alabama 35611

L. Raghaven, Project Manager
U.S. Nuclear Regulatory Commission
One White Flint, North
11555 Rockville Pike
Rockville, Maryland 20852

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

BROWNS FERRY NUCLEAR PLANT (BFN) - RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION (RAI) RELATING TO UNITS 2 AND 3 TECHNICAL SPECIFICATION (TS) CHANGE NO. TS-384 - POWER UPRATE OPERATION (TAC NOS. M99711 AND M99712)

This enclosure provides the TVA response to the May 7, 1998, NRC request for additional information.

NRC REQUEST 1

Tennessee Valley Authority's (TVA's) submittal for the power uprate amendment did not address several aspects of a design basis loss of coolant accident (LOCA). The staff is of the opinion that the following need to be considered in a design basis LOCA analysis.

(a) TVA assessed main steam isolation valve (MSIV) leakage with regard to the control room. The same release affects persons at or beyond the site boundary. Therefore, please address exclusion area boundary (EAB) and low population zone (LPZ) doses due to MSIV leakage.

(b) NUREG-0737 Item III.D.3.4 specifically states that the design basis LOCA source term should be consistent with Appendix A and B of Standard review Plan (SRP) Chapter 15.6.5. The SRP, Appendix B addresses emergency core cooling system (ECCS) leakage. Please provide an assessment of EAB, LPZ, and Control Room doses due to leakage from ECCS outside of primary containment. Otherwise, please provide a technical justification why TVA believes these credible release paths need not be addressed.

Please note that although the above issues may not be currently addressed in the Browns Ferry Nuclear Plant (BFN) Updated Final safety Analysis Report (UFSAR) accident analyses, the staff believes that they must be addressed to enable the staff to make a finding that public protection is not adversely affected by the proposed power upgrade.

TVA REPLY 1

a. The design and licensing bases of BFN do not include MSIV leakage into the turbine building in the EAB and LPZ dose

calculations. However, calculation ND-Q0031-920075, "Control Room Doses," Revision 7, Attachment 10, contains a General Electric (GE) calculation that includes values for the contribution of this leakage pathway. Even though the GE calculation contains values for these contributions, MSIV leakage into the turbine building is not part of the BFN licensing or design bases. The GE determined values due to MSIV leakage are as follows:

	Whole Body (REM)	Thyroid (REM)
LPZ	0.01	1.98
EAB	0.00	0.00

As can be seen from the above, these values are an insignificant contribution to the allowable 10 CFR 100 doses. These values were calculated in accordance with the Boiling Water Reactor Owners Group (BWROG) radiological dose calculation methodology for MSIV leakage.

b. By letters dated March 17, 1981, and July 27, 1982, in response to the requirements of the Task Action Plan as promulgated in NUREG-0737, TVA submitted their responses to Item III.D.3.4, "Control Room Habitability." In the March 17, response, TVA provided their reply to Item III.D.3.4 of NUREG-0737 which, based on the current licensing basis for BFN, did not include EAP, LPZ, and control room doses due to ECCS leakage outside of primary containment.

By letter dated August 30, 1982, NRC provided the safety evaluation detailing BFN's compliance with the NUREG requirement. Based on their review, NRC found that the current plant systems will provide safe, habitable conditions within the control room under both normal and accident radiation and toxic gas conditions including LOCAs. This did not include control room doses due to ECCS leakage from outside containment. NRC further concluded that the BFN design met the criteria identified in Item No. III.D.3.4 of NUREG-0737.

As stated above, affects of MSIV leakage and ECCS leakage from outside containment are not considered in the current BFN license basis. However, as described in the cover letter, TVA has proposed an operating license condition to provide an analysis of the design basis LOCA to confirm compliance with General Design Criteria (GDC)-19 and offsite dose limits considering MSIV leakage and ECCS leakage.

NRC REQUEST 2

While TVA is performing the requested ECCS leakage dose analysis outside the primary containment as requested in item 1 above, to facilitate our continued review, please provide a design basis value for the ECCS leakage (gpm) outside of the primary containment. The staff will multiply this value by 2 as provided for in the SRP.

TVA REPLY 2

Because ECCS leakage dose rates are not part of the licensing basis, there is no design basis value for ECCS leakage outside primary containment for BFN. (See the response to Request 1 for additional information.) However, the current value in Technical Specification 3.6.C for unidentified primary containment leakage is 5 gpm. This value would be conservative in its application to ECCS leakage outside primary containment and is substantially more than would be expected to occur.

As part of the analysis that will be performed in accordance with the proposed operating license condition provided in the cover letter, a more accurate value for ECCS leakage outside primary containment will be established.

NRC REQUEST 3

The control room infiltration at BFN is stated to be 3717 cfm. The filtered control room intake fans are rated at 3000 cfm. Does any of the 3000 cfm filtered intake displace any of the 3717 cfm unfiltered infiltration, i.e., is the net intake 717 cfm or 6717 cfm?

TVA REPLY 3

The net intake used in the control room dose calculation is 3717 cfm unfiltered and 3000 cfm filtered makeup, for a total net intake of 6717 cfm.

NRC REQUEST 4

The UFSAR describes the control room pressurization fans starting on a primary containment isolation signal or on high radiation signal. Please describe which signal is applicable for each of the analyzed accidents; LOCA, main steam line break (MSLB), rod drop accident (RDA), and fuel handling accident (FHA). If the isolation is based on a high radiation signal, please describe the basis of the setpoint and the expected time delay between the onset of the event and isolation of the control room. There have been recent 50.72 reports in several facilities that describe conditions under which the expected monitor alarm setpoints would not be reached for certain analyzed accidents potentially resulting in doses in excess of the General Design Criterion (GDC) 19 limits. If these data are not readily available, the Nuclear Regulatory Commission (NRC) staff will assume manual operator action after 30 minutes.

TVA REPLY 4

The control room emergency ventilation system (CREVS) is required to be operable for a LOCA, MSLB, FHA and RDA. CREVS is automatically initiated by a Group 6 primary containment isolation signal (PCIS) or high radiation at the control bay air intakes or can be manually initiated by the control room operators. The PCIS Group 6 is initiated by reactor vessel low water level, drywell high pressure or reactor building ventilation high radiation. The PCIS Group 6 CREVS initiation is anticipatory of the potential for adverse post accident control room conditions, whereas the control bay air intake radiation signal is a more direct indication of potential adverse control room conditions.

The setpoint for the control bay air intake radiation monitors is established based on an evaluation of the above list of potential events. The following describes the impact of each event on the setpoint.

For the RDA, FHA, and MSLB, control room operator dose evaluations have concluded that the operator doses will be below the allowable limits (10 CFR 50, Appendix A, GDC 19 which provides a limit of 5 rem whole body or 30 rem thyroid) even if the control room is not isolated. Therefore, the radiation monitor setpoint is not controlled by these events.

For a LOCA, the PCIS Group 6 initiation will occur significantly prior to the control room experiencing conditions which would result in excessive doses to the control room operators and, hence, significantly prior to an initiation on control bay air intake high radiation. The CREVS radiation monitor setpoint was determined assuming the PCIS Group 6 isolation did not occur.

Evaluations for the LOCA determined that the LOCA dose is spread over the entire 30 day event period (as opposed to approximately 2 hours for the other events) and results in a lower dose rate than the other events. The LOCA is the event that produces the limiting minimum dose rate.

The setpoint for the control bay air intake radiation monitors is set at a sufficiently low dose rate such that the operator dose will not exceed 30 rem during the course of a LOCA event. The

setpoint is based on the normal control bay air intake path and rate. The setpoint is based on the control room operators not exceeding the GDC 19 limits for any of the events even if the control room did not isolate prior to the control bay air intake radiation monitor signal. Therefore, it is the lowest dose rate, lowest activity concentration at the air intake, the lowest count rate which is the limiting case used to set the monitor.

The BFN Technical Specification allowable value (270 counts per minute) is within the calibratable range of the instruments.

NRC REQUEST 5

The power uprate was specifically requested for Units 2 and 3. TVA calculation ND-Q0031-920075 Rev. 7 (control room doses) explicitly discounts Unit 1 (sheet 7, 8 of 37). Accordingly, the staff's review will be limited to Units 2 and 3. However, the staff will consider the postulated doses to Unit 1 control room operators resulting from Units 2 and 3 design basis accidents. The control habitability issues as they apply to Unit 1 should be resolved prior to its operation. Please inform us of your schedule for revising the TVA calculation ND-Q0031-920075 Rev. 7 to address Unit 1 issues and providing the missing Unit 1 data for staff review.

TVA REPLY 5

Sheets 7 and 8 of the referenced calculation refer to mixing volumes and restricts the location of the accident to Units 2 and 3 only. The actual operator dose contained in the referenced calculation is applicable for all three unit control rooms for a Unit 2 or 3 accident. The Unit 1 and 2 control rooms are shared in a common room with Unit 1 at one end and Unit 2 at the other. The Unit 3 control room, though separated from the Unit 1/2 control room, is part of the same control bay habitability zone. The calculation does not address Unit 1 operation (and, therefore, mixing volumes for the Unit 1 reactor building) since the unit is in an extended shutdown with the fuel removed from the reactor vessel.

NRC REQUEST 6

TVA analysis ND-Q0031-920075 Rev. 7 (control room doses) indicates on sheet 13 of 38 that the turbine building volume of 2,100,000 cubic feet was used as a dilution volume for MSIV leakage. Data items d and e on page 14.6-29 of the UFSAR imply that dilution credit was taken. The staff does not consider this assumption acceptable in a design basis calculation. The condenser leak could occur at a single location. Convective forces will cause the release to ascend to the nearest roof vent.

These analysis descriptions contradict the information in the referenced (GE) report NEDC-32091 dated August 1992, which states on page B-14 that "for MSIV calculations, no credit has been taken for the turbine building...." While the turbine building volume is entered, the purge rate is set to 1E6 %/day, which establishes a near-instantaneous release to the environment. The computer input sheets in the GE letter to TVA, dated August 28, 1992, indicate that credit was effectively not taken.

Please update ND-Q0031-920075 Rev. 7 and the UFSAR to be consistent with the supporting calculation, such that the BFN design basis does not credit turbine building dilution for MSIV leakage.

TVA REPLY 6

As noted on sheet 12 of the referenced calculation, assumptions 26 through 33 are not used in the calculation since the BWROG methodology was used in place of the TVA modeling of MSIV leakage. The purge rate of 1.0E6 %/day in the GE letter of August 28, 1992 is the value used in the calculation of the MSIV leakage contribution to control room dose, which, as noted by the NRC, results in a near instantaneous release to the environment (i.e., essentially no dilution).

NRC REQUEST 7

ND-Q0031-920075 Rev. 7 sheet 12 of 38, states that TVA considers that all piping from the MSIVs to the condenser remain intact even though it is not seismically qualified. The calculation notes that: "This appears to be in accordance with the Boiling Water Reactor Owners Group (BWROG) position as well as others in industry." No further justification is provided. However, the GE NEDC-31858P report addresses the need to justify integrity of the main steam lines following a seismic event. The NRC has approved licensee applications of the GE methodology as a basis of increasing MSIV leakage. Each of these requests addressed these integrity concerns. Please provide a technical justification for your proposed approach or re-analyze the doses without using the GE BWROG methodology.

TVA REPLY 7

TVA has identified modifications that need to be implemented in order meet the requirements in GE NEDC-31858P. TVA has scheduled to implement these modifications during the refueling outages scheduled for Fall of 1998 on Unit 3 and Spring of 1999 for Unit 2.

As stated in the cover letter, TVA has proposed an operating license condition to provide an analysis of the design basis LOCA to confirm compliance with GDC-19 and offsite doses limits considering MSIV leakage and ECCS leakage. This analysis will include appropriate calculation of the doses based on the BWROG approach to address the piping from the MSIVs to the condenser.

NRC REQUEST 8

TVA's submittal of October 1, 1997 and letter dated April 1, 1998 described the methodology for analyzing the radiological impacts due to the proposed power uprate. This methodology involved a scaling factor based on the change in power level that was then applied to the results from prior analyses. Reactor core inventories were recalculated using a different methodology than that used in the earlier licensing analyses. In reviewing your submittal, the NRC staff has identified apparent discrepancies between the TVA's proposal and the existing UFSAR analyses. For example:

a. Table 3 on Page 8 of the environmental assessment (EA) for the proposed uprate provides results for the MSLB analysis and contains the text: "Iodine concentration in coolant = 26 μCi/g dose equivalent I-1 31." Section 14.6.5.2.1.b of BFN-15 tabulates the iodine concentrations assumed in the current analyses. The existing analysis assumptions are not based on 26 μCi/g.

b. Table 3 reports that the pre-uprate analysis result for the exclusion area boundary was 0.66 rem whole body and 32.1 rem thyroid. Contrary to this, Section 14.6.5.3 of the UFSAR reports 0.0012 rem whole body and 0.65 rem thyroid. Table 14.9-1 of the UFSAR provides EAB results of 0.017 rem whole body and 2.9 rem thyroid.

c. Table 5 on Page 9 of the EA for the proposed uprate provides pre-update EAB doses for the RDA analysis as 0.055 rem whole body and 1.62 rem thyroid. Table 14.6-2 of the UFSAR provides limiting values of 0.0056 rem whole body and 2.4E-4 rem thyroid. Table 14.9-1 of the UFSAR provides EAB results of 0.012 rem whole body and 6.1 rem thyroid.

d. Table 4 on Page 8 of the EA for the proposed uprate provides results for the FHA analysis and contains the text: "Fuel Handling Accident (single fuel bundle and handling equipment dropped)." The assumptions in Section 14.6.4.1 do not appear to address the weight of the handling equipment.

e. Section 14.6 of the UFSAR does not report numerical values for the analysis results for the LOCA and FHA. Table 14.9-1 does provide numeric results. The data in the corresponding tables in the environmental assessment are not consistent with the Table 14.9-1 results.

f. Section 14.6.2.8.4 and TVA letter dated April 4, 1994, address release paths for the RDA that do not appear to be reflected in the power uprate submittal. The April 4, 1994 letter postulated a thyroid dose of 18.1 rem. This is not consistent with the power uprate submittal which indicates the present (pre-uprate) thyroid dose to be 1.62 rem.

Therefore, please:

i. explain why the pre-uprate data in the submittal is not consistent with the current design basis as described in the UFSAR.

ii. provide a certification that the post-uprate data in the submittal is based on the current design basis as stated in the UFSAR, modified only by those factors explicitly identified in the submittal.

iii. provide a tabulation, in a format similar to that of Table 15-4 of Regulatory Guide 1.70, the analysis assumptions and parameters that will constitute the design basis of Browns Ferry Units 2 and 3, following the uprate, with regard to the Chapter 14 design basis radiological accident consequences to persons offsite and in the control room.

<u>TVA REPLY 8</u>

i. The October 1, 1997 and the April 1, 1998 submittals are based on the latest BFN calculations. BFN recognizes the importance of accurately and comprehensively updating the UFSAR to reflect changes to the documentation and to the plant. BFN has also recognized that in general, the UFSAR is not current and, thus, may not reflect the latest calculations or plant information. This issue is being tracked by two separate corrective action program documents and is in the process of being rectified. It is planned to update the radiological dose information in the UFSAR as part of the power uprate project.

The NRC request identified several locations where the values are not consistent with the UFSAR. As discussed above, each of these will be corrected; however, the following provides an explanation for each deviation.

a. The MSLB dose calculation is based upon a value (32 µCi/gm) which is conservative compared to the current Technical Specification (TS) 3.6.B.6 which allows activity concentrations in the reactor coolant to be as high as 26 µCi/gm of dose equivalent I-131 for up to 48 hours following a power transient. Both of these values are conservative compared to the values in the UFSAR.

b. The differences noted by the NRC between the submittals and UFSAR Section 14.6.5.3 result from the UFSAR not reflecting the latest calculations as discussed above. As stated in the introduction paragraph of Section 14.9, this section (and the associated following sections) of the BFN UFSAR are considered historical and are not kept up to date. They are retained in the UFSAR as background information only. The official results of the dose calculations, once corrected as discussed above, will reside in Section 14.6 of the UFSAR.

c. The differences noted by the NRC between the submittals and UFSAR Table 14.6-2 result from the UFSAR not reflecting the latest calculations as discussed above. As stated in the introduction paragraph of Section 14.9, this section (and the associated following sections) of the BFN UFSAR are considered historical and are not kept up to date. They are retained in the UFSAR as background information only. The official results of the dose calculations, once corrected as discussed above, will reside in Section 14.6 of the UFSAR.

d. The inclusion of the words "and handling equipment" is correct. As discussed in BFN UFSAR Section 14.6.4, the current General Electric BWR analysis of a FHA is described in the licensing topical report for nuclear fuel, "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A, Section S.2.2.3.5 and subsequent revisions thereto. The analysis is based on the dropping of a single fuel bundle and the fuel grapple. BFN does not have the heavier NF-500 refueling mast mentioned in the GE document and, therefore, the analysis bounds BFN. The fuel grapple is assumed to drop due to the fuel grapple main hoist cable breaking as discussed in BFN UFSAR Section 14.6.4.1 Item 2.

e. As stated in the introduction paragraph of Section 14.9, this section (and the associated following sections) of the BFN UFSAR are considered historical and are not kept up to date. They are retained in the UFSAR as background information only. The official results of the dose calculations, once corrected as discussed above, will reside in Section 14.6 of the UFSAR.

f. The October 1, 1997 submittal contained incorrect values for the Control Rod Drop Accident in the Environmental Assessment and Table 9-6 of the safety evaluation. The values were incorrectly extracted from the supporting calculations. Attached are revised pages. The revised pages are also consistent with the information contained in the April 4, 1994 submittal.

Following the identification of this discrepancy, the other dose values provided in the October 1, 1997 submittal have been reviewed and verified to be correct.

ii The values contained in the power uprate submittal reflect the latest BFN calculations; however, as stated in the cover letter, TVA has proposed an operating license condition to provide an analysis of the design basis LOCA to determine the resulting control room and offsite doses considering MSIV leakage and ECCS leakage. The results of these calculations will be used to update the UFSAR as described in Item i.

iii. The following table provides the tabulation similar to that of Table 15-4 of Regulatory Guide 1.70 of analysis assumptions and parameters that will constitute the design and licensing basis of Browns Ferry following the uprate.

BFN Comparison to Table 15-4 of Regulatory Guide 1.70

It should be noted that the below listed values are based on current dose calculations. As a result of recent discussions, TVA will be revising control room and off site dose calculations to incorporate updated X/Q values and to include the dose due to MSIV and ECCS leakage into the above BFN dose calculations. During this review and update, some of the below values may be subject to change.

1. Loss of Coolant Accident

   a. Hydrogen Purge Analysis

      N/A for BFN. Venting is assumed to occur for a period of 24 hours at a flow rate of 139 cfm at 10, 20, and 29 days post accident. This results in a discharge flow of equal amounts through the standby gas treatment system (SGTS).

   b. Equipment Leakage Contribution to LOCA Dose (NOTE: Not part of current licensing basis)

      (1) Iodine concentration in sump water after LOCA - 25% of core inventory.
      (2) Maximum operational leak rate through pump seals, flanges, valves, etc. - Undetermined at this time.
      (3) Maximum leakage assuming failure and subsequent isolation of a component seal - Undetermined at this time.
      (4) Total leakage for (2) and (3) above - Undetermined at this time.
      (5) Temperature of sump water vs. time - < 177° F at all times.
      (6) Time intervals for automatic and operator action - N/A
      (7) Leak paths from point of seal or valve leakage to the environment - All leakage is to the reactor building and then through SGTS to the environment.
      (8) Iodine partition factor for sump water - 10 at all times.
      (9) Charcoal adsorber efficiency assumed for iodine removal - 90%.

   c. Main Steam Line Isolation Valve Leakage Control System Contribution to LOCA Dose (BWR)

      N/A for BFN. BFN does not have a MSIV Leakage Control System.

3. Main Steam Line and Steam Generator Tube Failure

a. N/A for BWRs
b. Potential increase in iodine release rate above the equilibrium value - 32 µCi/gm - factor of ten increase above normal Technical Specification allowable maximum.
c. Chronological list of system response times, operator actions, valve closure times, etc. - The release occurs in 5.5 sec. All of the release is released to the environment from the turbine building within 2 hours of the event.
d. Steam and water release quantities - Mass of steam released = 1.987E4 lbm; Mass of water released = 4.374E4 lbm (38% is assumed to flash to steam).
e. Iodine transport mechanisms and release paths - The steam portion and the water portion that flashes to steam (38%) is completely released to the turbine building and from there to the environment. The water portion that does not flash to steam is not released. 50% of the released iodines are also assumed to plate out on turbine building surfaces.
f. N/A for BFN
g. N/A for BFN

4. Fuel Handling Accident

a. Number of fuel rods in core - 48132.
b. Number, burnup, and decay time of fuel rods assumed to be damaged - 125 fuel rods (8x8 design) are assumed to be damaged. The TID 14844 methodology at 1000 effective full power days (EFPD), 3458 MWt and a 24 hour decay time is used to determine the source term released.
c. Radial peaking factor - 1.5.
d. Earliest time after shutdown that fuel handling occurs - 24 hours.
e. Amounts of iodines and noble gases released into pool - 10% of iodines and noble gases are released, with the exception of 30% of Kr-85 released .
f. Pool decontamination factors - 133 for inorganic iodine and 1 for organic iodine in which 0.25% is assumed to be organic and 99.75% is assumed to be inorganic.
g. Time required to automatically switch from normal operation - Within 15 seconds the normal ventilation path is isolated and the remainder of the release is through the SGTS system.
h. Amount of release not routed through ESF-grade filters - 5,500 ft^3.

5. <u>Control Rod Drop Accident</u>

a. Percent of rods undergoing clad failure - 850 rods out of 48132 (8x8 fuel) fail.
b. Radial peaking factor - 1.5.
c. Percent of fuel reaching melting temperature - Mass fraction of fuel = 0.0077.
d. Peaking factors - 1.5.
e. Percent of core fission products assumed released into reactor coolant - For the non-melted fuel - 10% of noble gases and iodines are released. For the melted fuel, 100% of the noble gases and 50% of the iodines are released.
f. N/A for BFN
g&h. Summary of containment system parameters - A condenser volume of 187,000 ft^3, mechanical vacuum pump flow of 1850 cfm, 10 cfm base of stack leakage , and 90% iodine plate out in the condenser were used.

Additionally, failure of the non-safety related pressure protection on the recirculation sample line causes a failure of the downstream piping. The piping failure allows water to enter the reactor building. The secondary containment is isolated and SGTS initiated at 7 minutes based on high radiation. The isolation is based on exceeding the Technical Specification limit of 100 mR/hr. Prior to isolation, release is directly from the reactor building at ground level. The following details are also provided:

- Reactor zone volume = 1,335,000 ft^3 (single reactor zone prior to secondary containment isolation)
- Air flow from the reactor zone prior to isolation = 95,000 cfm
- 100% of iodine is available for release to the reactor building
- SGTS flow = 22,000 cfm
- Secondary containment volume = 1,931,500 ft^3
- Iodine mixed within 26,500 ft^3 of reactor coolant
- Failed sample line releases 220 ft^3/hr (10651 lbm/hr)
- Flashing fraction = 36%

NRC REQUEST 9

GE Nuclear Energy, "Generic Guidelines for General Electric Boiling Water Reactor Power Uprate," Licensing Topical Report, NEDC-31897P-A, May 1992, Section 5.11.9, "Power Uprate Testing" indicates that performance testing will be conducted for systems and components which have revised performance requirements. Please identify the systems and components that will be tested.

TVA REPLY 9

A test specification is being prepared that identifies the performance tests and associated acceptance criteria recommended by GE during the ascension to power uprate conditions. This document is the result of a test selection process that is based on the BFN original startup test specification and previous GE BWR power uprate test programs.

Examples of plant systems considered for performance testing to demonstrate revised performance requirements include:

- Intermediate Range Neutron Monitoring
- Average Power Range Monitoring
- Reactor Core Isolation Cooling
- High Pressure Coolant Injection
- Standby Liquid Control
- Reactor Water Level Measurement
- Turbine Generator
- Electro-Hydraulic Pressure Control
- Feedwater Control
- Reactor Recirculation
- Drywell Atmosphere Cooling

NRC REQUEST 10

Please describe how preoperational testing will be conducted in accordance with the requirements of 10 CFR 50, Appendix B, Criterion XI, "Test Control."

TVA REPLY 10

The tests performed during power ascension to power uprate conditions will be performed in accordance with the existing BFN Test Program (SSP 8.1, "Conduct of Testing"), using BFN test procedures. This test program meets the requirements of 10 CFR 50, Appendix B, Criterion XI.

REFERENCES:

1. TVA letter to NRC dated October 1, 1997, Browns Ferry Nuclear Plant (BFN) - Units 2 and 3 - Technical Specification (TS) Change TS-384 - Request For License Amendment for Power Uprate Operation

2. TVA letter to NRC dated March 16, 1998, Browns Ferry Nuclear Plant (BFN) - Units 2 and 3 Technical Specification (TS) No. 384 Supplement 1 - Request for License Amendment for Power Uprate Operation

3. TVA letter to NRC dated March 20, 1998, Browns Ferry Nuclear Plant (BFN) - Units 2 and 3 - Technical Specification (TS) Change TS-384 - Request for License Amendment for Power Uprate Operation

4. TVA letter to NRC dated April 1, 1998, Browns Ferry Nuclear Plant (BFN) - Response to Request for Additional Information (RAI) Regarding Units 2 and 3 Technical Specification (TS) Change TS - 384, - Request for License Amendment for Power Uprate Operation, (TAC Nos. M99711, M99712) and Resolution of Control Room Emergency Ventilation System (CREVS) Issues (TAC Nos. M83348, M83349, M83350)

5. NRC letter to TVA dated May 7, 1998, Browns Ferry Plant Units 2, and 3: Request for Additional Information Relating to Technical Specification Change No. TS-384 - Power Uprate Operation (TAC Nos. M99711 and M99712)

6. TVA letter to NRC dated March 17, 1981, Browns Ferry Units 1, 2, and 3, Response to Item III.D.3.4, Control Room Habitability Requirements

7. TVA letter to NRC dated July, 27, 1982, Browns Ferry Units 1, 2, and 3, Additional information related to NUREG-0737, Item III.D.3.4, Control Room Habitability

8. NRC letter to TVA dated August 30, 1982, in regards to Browns Ferry Units 1, 2, and 3, NUREG-0737, Item III.D.3.4, - Control Room Habitability

9. TVA Letter to NRC dated April 4, 1994, Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 - Technical Specification (RS) No. 322, Revision 1 - Elimination of Main Steam Line Radiation Monitor (MSLRM) Scram and Isolation Functions

ENCLOSURE 2
TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT
UNITS 2 AND 3

BROWNS FERRY NUCLEAR PLANT (BFN) - RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION (RAI) RELATING TO UNITS 2 AND 3 TECHNICAL SPECIFICATION (TS) CHANGE NO. TS-384 - POWER UPRATE OPERATION (TAC NOS. M99711 AND M99712)

REVISED PAGES TO OCTOBER 1, 1997 SUBMITTAL

The following are revised pages to the Environmental Assessment and Table 9-6 of the safety evaluation of the October 1, 1997 submittal regarding power uprate.

Table 5

CRDA Radiological Consequences

	3293 MWt	3458MWt	
LOCATION	DOSE	DOSE	LIMIT
Exclusion Area:			
Whole Body Dose, rem	2.43	1.43	25
Thyroid Dose, rem	18.1	19.0	300
Low Population Zone:			
Whole Body Dose, rem	1.37	0.97	25
Thyroid Dose, rem	12.6	13.99	300

The plant specific results shown in the above tables for the power uprate project remain well below established regulatory limits.

3.4 Other Environmental Impacts

There should be no effects on groundwater, erosion, sediment and stormwater, wastewater, cultural, archaeological, and historical resources, land use, noise, solid and hazardous waste, aesthetics, socioeconomics, or traffic as a result of this project. There would be no nonradiological permits modified nor and new permits generated as a result of this project.

4.0 AGENCIES/PERSONS CONSULTED

The information presented in this document was prepared with input from various internal TVA organizations and by TVA's contracting partner for this project. No outside agencies or persons were consulted.

Table 9-6

CRDA Radiological Consequences

LOCATION	Pre-uprate DOSE	Uprated DOSE	Dose LIMIT
Exclusion Area:			
Whole Body Dose, rem	2.43	1.43	25
Thyroid Dose, rem	18.1	19.0	300
Low Population Zone:			
Whole Body Dose, rem	1.37	0.97	25
Thyroid Dose, rem	12.6	13.99	300

TEA:DAH:BAB
Enclosures
cc (w/o Enclosures):

M. J. Burzynski, BR 4X-C
E. S. Christenbury, ET 11H-K
C. M. Crane, PAB 1E-BFN
K. N. Harris, LP 6A-C
F. C. Mashburn, BR 4X-C
D. T. Nye, PEC 2A-BFN
Dale Porter, MOD 2A-BFN
C. M. Root, PAB 1G-BFN
J. A. Scalice, LP 6A-C
K. W. Singer, POB 2C-BFN
NSRB Support, LP 5M-C
EDMS, WT 3B-K

q:lic/submit/techspecs/ts384puo.doc