TVA

Tennessee Valley Authority, Post Office Box 2000, Decatur, Alabama 35609-2000

April 1, 1998

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

Gentlemen:

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

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)

This letter provides additional information requested by NRC in support of TS-384, and the resolution of CREV system issues. 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. Also on July 31, 1992, TVA provided a letter describing corrective actions resolving previous deficiencies identified with the CREVS.

Enclosure 1 provides TVA's response to the February 18, 1998, NRC RAI for both the October 1, 1997, proposed TS change, and the July 31, 1992, CREVS issue letter. This letter includes replies to all requests except B.2.

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In Request B.2., NRC requested that TVA explain the methodology used to determine the control room atmospheric dispersion (X/Q) values used in dose analysis. That is, justify the use of the closest control room ventilation intake point or re-assess the X/Q values used. TVA is re-calculating the X/Q values for non fumigation top of the stack releases considering both potential control room ventilation intake points. If necessary, TVA will revise the control room dose calculation, and will provide the calculation, and its effect, if any, on the control room dose, in a supplement to this RAI response.

The commitment made in this letter is contained in Enclosure 2. If you have any questions, please telephone me at (256) 729-2636.

Sincerely,

T E Abney

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ENCLOSURE 1

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

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION 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)

This enclosure provides TVA's response to the February 18, 1998, NRC Request for Additional Information.

A. **TECHNICAL SPECIFICATION CHANGE TS-384 REQUEST FOR LICENSE AMENDMENT FOR POWER UPRATE OPERATION (TAC NOS. M99711. M99712)**

The October 1, 1997, submittal describes the process by which accident doses determined at the previous power level were ratioed to estimate the doses at the proposed power level. The staff recognizes that the increase in doses would normally be proportional to the increase in thermal power or the increase in rated steam flow. The submittal does not, however, provide sufficient information for the staff to make the findings necessary to approve the amendment. Please refer to the radiological analysis discussions in the NRC Safety Evaluation Reports issued on the General Electric (GE) topical reports. To facilitate the review, please provide the following information:

NRC Request A.1.a.

A description of the assumptions, inputs and methodology used in each analysis. The Final Safety Analysis Report (FSAR) descriptions, generally do not provide sufficient detail for the staff to assess the acceptability of assumptions, inputs, and methodologies used.

TVA Reply

<u>Purpose of Evaluation</u>

The purpose of the power uprate radiological evaluation is to quantify the increased radiation levels in order to determine if the radiation safety criteria can still be met for BFN at the power uprate condition. For this purpose, the radiation criteria are: the regulatory requirements 10 CFR 100 and General Design Criteria (GDC)-19 relating to the radiation design basis Loss of Coolant Accident (LOCA), Control Rod Drop Accident (CRDA), Fuel Handling Accident (FHA), and Main Steam Line Break Accident (MSLBA); the requirements of 10 CFR 50 Appendix I and 40 CFR 190 for radiation levels offsite during normal operation and the requirements of 10 CFR 20 for onsite workers. The quantified sources are also needed to demonstrate that the electrical equipment can still be qualified under the requirements of 10 CFR 50.49.

Scope of Evaluation

The scope of this review is to assess the change in radiation environment from a 5 percent increase in power during plant operation. The review also considers the change in radiation source terms. Source terms for this purpose include direct radiation from the core, accumulation of fission products in the core, fission products transported by water or steam that are "leaked" from the core, and activation of isotopes that pass through the core. The change in the source terms is used to evaluate the changes in radiation environments. The expected changes in the dose rates were estimated/scaled by the changes in sources. This approach is consistent with the regulatory guidance provided by NEDC-31897P-A "Generic Guidelines for General Electric Boiling Water Reactor Power Uprate", May 1992, specifically Section 5.4, Appendix H.

The process of evaluation has two parts. In the first part, the source term change is evaluated. In the reactor core, the radiation source term was assumed to be proportional with power. In the second part, the effect of the change on existing TVA radiation calculations is evaluated. The emphasis of this response is on the change in the radiation sources and the impact of source term changes on the pre-uprated radiation dose calculations. The evaluation of the design basis radiation sources was performed by recalculating the sources for uprate power conditions. The revised sources are compared to the existing sources to identify the changes utilizing the methodology described in NEDC-31897P-A. If the new source term values resulted in radiation doses which were lower than the existing radiation doses, then the existing radiation doses were retained. Otherwise, the existing radiation dose values were increased by the appropriate scaling factors based on the new source term values.

Evaluation of the change in sources is nominally a direct comparison, for example, the radiation leaving the core is 5 percent greater than the radiation leaving the pre-uprated core. The exception to this is the evaluation of the fission product inventory. For power uprate, the fission product inventory was changed by core power, irradiation time and methodology of calculation (as discussed below). To evaluate these conditions, the isotopic inventory was converted to infinite air dose for each isotope. The total dose including isotopic decay was integrated over two hours, 30 days and 100 days. The ratio of the integrated dose for the uprate inventory to the pre-uprate inventory becomes a measure of change between the existing and updated inventories. The ratios formed for the inventories included the noble gasses, iodine and solids. These values plus the direct comparison ratios can be used to scale or estimate radiation levels in the plant.

Baseline Used for Power Uprate Evaluation

Starting from the BFN design basis calculations (TID-14844), the BFN fission product inventory has been re-calculated using the computer program ORIGEN for power uprate. The methodology change was required in order to accommodate the planned 1400 effective full power days (EFPD) exposure associated with a 24 month fuel cycle. The case selected was based on a 170 kg bundle with 4.1 percent enrichment and operated for 1400 EFPD at 3458 MWt. The 1400 EFPD, 170 kg bundle case is bounding for the expected fuel bundle to be used for power uprate as was shown by sensitivity studies varying the bundle weight and enrichment and the irradiation method.

Evaluation Results

When the uprate source term inventory is compared to the pre-uprate inventory, the noble gas inventory decreases. This is because the pre-uprate inventory which was based on pre-uprate methodology did not allow for neutron capture by fission products. However, the iodine inventories in the uprate source terms did increase.

The radiation source in the reactor core is proportional to the fission rate. Therefore, the total source in the core increases by 5 percent. The direct radiation leaving the core is influenced by the fraction of the total power in the edge bundles. For the uprate analysis the increase in vessel flux was conservatively assumed to be 16 percent.

The indirect radiation sources in the plant are the result of "imperfections/leaks" of fission products from the fuel or activation of isotopes passing through the core, for example, corrosion products. The fission product concentrations in the steam and water become design basis concentrations. Design basis data contain a contingency for operational periods when the release levels in the fuel are higher than currently observed.

The coolant activation concentrations were evaluated by recalculating the activity of N-16. The revised computation included the power change plus the pressure and flow changes. Traditionally, C-15 is also treated as part of N-16 because of the great difficulty in distinguishing between N-16 and C-15. For uprate conditions the N-16 activity increases 5 percent in the volatile nitrogen compounds in the reactor steam. Since the total steam flow to the turbines also increases by 5 to 6 percent for uprate conditions, the activity per unit mass in the steam is approximately constant.

It is postulated that the corrosion product activities could increase by more than 5 percent. This is because the inflow to the reactor of isotopes available to be activated is constant and the activating neutron flux increases 5 percent. Since the feedwater flow increases 5 percent, this is effectively a 5 percent flow increase and a 5 percent power increase. This is potentially an increase to 110 percent. However, the power uprate calculations based on American Nuclear Society-18.1 Radioactive Source Term Standard For Normal Operation Of Light Water Reactors, indicate no increase in corrosion product activity due to increased corrosion product removal by the condensate demineralizer system.

Nothing has been identified in the source evaluation leading to a large unacceptable change in the dose levels. Given the relationship between power and radiation release, increases are expected. This is particularly true for radiation doses which have been calculated based on dose rate surveys or measured environmental releases. In the area of in-plant radiation levels, the dose surveys can be expected to increase with power uprate. Increases of a few percent do not impact the operation of the plant. Industry-wide occupational radiation exposure reports for commercial nuclear power reactors indicate that most of plant exposure has been shown to be the result of maintenance activities performed during plant outages. Therefore, the total occupational radiation exposure for commercial nuclear power reactors is generally not dependent on reactor power.

The radiation dose used for equipment qualification is the integrated normal operating dose plus the maximum accident dose along with the required percentage safety margin which is independent of power level. The integrated normal operating dose is based on in-plant radiation survey data. Portions of the normal operating dose will increase approximately in proportion to the power increase. Other portions of the normal operating dose, such as crud deposits, will not change.

The post accident equipment qualification dose results from fission product isotopes released to circulating reactor water or fission products becoming airborne during an accident. As stated above for uprate source terms, the inventory of noble gases available to be released decreases due to the transition to the ORIGEN methodology. This causes some of the post accident doses to decrease rather than increase. The net result of the power uprate is a small change (≤5 percent) to some of the radiation environments. A review of the data contained in the BFN equipment qualification files indicates that there will be no adverse impact on equipment qualification from the change in integrated normal plus accident doses due to a 5 percent increase in power.

The design basis accident offsite dose analysis for the four events (LOCA, CRDA, FHA and MSLBA) changes as a result of the fission product inventory changes (due to both the power increase and the transition to the ORIGEN methodology). Using the 105 percent power, 1400 day inventory, the offsite gamma dose will be reduced; however, the thyroid dose will increase (≤5 percent). The new dose levels still meet the requirements of the regulatory guides and the standard review plans.

Ratios of radiation conditions with and without power uprate are supplied in Table 1. This table includes both those radiation levels that will increase as well as those that will remain unchanged. The increased radiation levels are acceptable because the levels either still meet the requirements and/or can be managed through plant operating/maintenance practices.

Conclusion

With plant operation at power uprate conditions and the associated increased radiation levels, the safety criteria continue to be met at BFN. The scaled up doses indicate a small increase in the thyroid doses and a small decline in whole body doses. The radiation sources in the plant contributing to the offsite dose are a small fraction of the design basis sources in the reactor coolant offgas. These sources will increase proportional to power increase but will remain below regulatory requirements.

Table 1

NRC Request A.1.a
Summary of Power Uprate
Scale Factors

Normal Operation	Scale Factor
Reactor Core	1.05
Vessel Flux	1.16
Off-Gas Rates	No Change
Reactor Coolant Fission Products	No Change
Activation Products	No Change
N-16	1.05
Reactor Steam N-16	1.05
Turbine Bldg N-16	1.05
Radwaste Activities	1.05

Design Basis Accident	Scale Factor		
	2 hr	30 days	100 days
Noble Gas	0.65	0.77	*
Released Solids	0.86	0.97	1.01
Iodine	1.05	1.1	*

* The 100 day doses are not required to analyze design basis events in accordance with NRC Regulatory Guide 1.3 of NUREG 0800.

NRC Request A.1.b.

A technical justification for any significant deviations from analysis guidance in applicable regulatory guides and Standard Review Plan (SRP) chapters (15.X.X).

Note: While the staff may perform confirmatory calculations, the staff's finding that offsite and control room doses are acceptable must be based on the licensee's design analysis. This is necessary for maintaining the plant's design basis. The staff must review the licensee's assumptions, inputs, and methodologies in making these findings. While submittal of the actual analysis is preferable, the staff recognizes that these analyses may be considered proprietary, and will accept tabular summaries if this information. Information in the format and content shown in Table 15.4 of Regulatory Guide 1.70 would be adequate.

TVA Reply

New radiological calculations were not performed for uprated conditions. The pre-uprated radiological calculations were evaluated and where necessary, the results of these calculations were modified for uprated conditions based on scaling factors which were determined in accordance with the methodology described by NEDC-31897P-A. The radiological evaluation performed for the Power Uprate Technical Specification change followed regulatory guidance provided in NEDC-31897P-A, "Generic Guidelines of General Electric Boiling Water Reactor Power Uprate", May 1992, specifically Section 5.4 and Appendix H.

NRC Request A.2

The October 1, 1997, power uprate submittal did not address the impact of the increased power level on the radiological consequences of postulated accidents to the control room operators. The staff must make a finding that the postulated operator doses will continue to comply with 10 CFR part 50, Appendix A, GDC-19, as clarified in SRP Chapter 6.4, for all accidents. Please submit a description of the assumptions, inputs, and methodologies used; and the obtained results of the TVA re-analysis of the control room doses. For events that do not result in a primary containment isolation signal, please assess the impact of delays in reaching the radiation monitor alarm set point, or time to complete operator manual isolation.

TVA Reply

The BFN power uprate evaluation for radiological impact considered the impact of the increased core thermal power on the radiological consequences of postulated LOCA and Main Steam Line Break accidents to the control room operators. This is shown in Section 9-2 and Tables 9-3 and 9-4 of the plant-specific safety analysis power uprate licensing report, October 1, 1997 letter. The control room dose analysis for power uprate was based on the most recent revision of BFN calculation ND-Q0031-920075, "Control Room Doses," Revision 7, April 1996, and were calculated using the scaling factor approach. The assumptions and methodology used for the determination of the scaling factors to be applied to the pre-uprate radiological calculation results are described in NEDC 31897P-A, Appendix H. For a more detailed discussion of the approach used in the development of the scaling factors, please refer to the reply to Comment A.1. above.

B. **CONTROL ROOM EMERGENCY VENTILATION SYSTEM CORRECTIVE ACTIONS**

NRC Request B.1.

By letter dated July 31, 1992, TVA provided a corrective action plan to address control room habitability concerns. This plan is still under review. Please provide a summary of the status of the items identified in the plan, e.g., which actions have been implemented and which ones are pending. In addition, the staff previously was provided copies of the following analyses.

Control Room Doses, RIMS R14 92 0903 110, including attachments.

Control Room X/Q, RIMS R14 92 0727 105

Control Room Doses from MSIV, RIMS R92 920904 001

If these analyses have been updated, please provide copies (or as a minimum, a description) of the of the revision(s) to facilitate the staff's review.

TVA Reply

The corrective actions specified in Enclosure 2 of the July 31, 1992, letter are complete.

TVA Calculation ND-Q0031-920075 "Control Room Doses," RIMS R14 92 0903 110, has been revised. The other calculations ha e not been revised. A copy of the revised calculation was provided to NRC on February 25, 1998.

NRC Request B.2.

TVA's letter dated August 10, 1994, explained the methodology used to determine the control room atmospheric dispersion (X/Q) values used in the dose analysis. The staff has a concern with the extremely low values of X/Q postulated for an elevated release. Our confirmatory analyses indicate that the postulated values may be low by as much as six orders of magnitude. In the analysis, the distance to the closest Control Room Ventilation System (CREV) intake (i.e., Unit 1) was used. The Unit 3 intake, although farther away, would appear to yield the most restrictive, X/Q value. For an elevated release, the ground level concentrations increase rapidly with the increasing distance until the lower surface of the plume reaches ground level due to vertical diffusion. From this point, the concentrations decrease with increased distance. Both CREVS intakes are within this cavity where the concentrations are increasing. The concentrations due to fumigation may increase similarly. Please justify the values or re-assess the X/Q valves used and update the dose analysis accordingly.

TVA Reply

As noted by the RAI question, the selection of the nearest control room ventilation intake point from the stack may result in non-conservative X/Q values, as compared to the more distant control room ventilation intake point. TVA acknowledges this non-conservative condition for the non-fumigation top of stack release. However, the X/Q value for the fumigation top of stack release, per the equation identified in Regulatory Guide 1.145, and the X/Q for the bottom of the stack release, should each be larger at the nearest control room ventilation intake. Since the bottom of the stack release X/Q is more than 10 orders of magnitude larger that the non-fumigation top of stack X/Q, the aforementioned non-conservatism may not materially affect the 2-hour, 8-hour, 16-hour, 3-day, and 26-day control room doses.

To ensure that the most conservative results are being evaluated, TVA is recalculating the X/Q values for the non-fumigation top of stack releases at both control room ventilation intake points. If necessary, TVA will revise the control room dose calculation, and will provide the calculation, and its effect, if any, on the control room dose, in a supplement to this RAI response submittal.

NRC Request B.3.

The control room corrective action plan describes the installation of dual CREVS intakes on either side of the turbine building. FSAR section 10.12.5.3 describes the intakes as being from the ventilation towers. This is also shown on FSAR figure 10.12-2b. FSAR Section 14.6.3.6 describes the configuration as described in the corrective action plan. Please explain the differences in these system descriptions.

TVA Reply

FSAR section 10.12.5.3 Control Building Subsection "Control Building HVAC", paragraph 4 on page 10.12-6 is correct as written. That is "Outside air for CREVS is drawn from both the main outside air intake ducts supplying ventilation tower 1 and ventilation tower 3. Outside air pulled from these intakes passes through a HEPA filter bank located in ventilation tower 2."

Figure 10.12-2a of the FSAR depicts the single HVAC duct shown on FSAR figure 10.12-2b which is supplied by two intake structures.

NRC Request B.4.

FSAR Figure 10.12.2a implied that the previous auxiliary pressurization fans (fans 31-151, 31-153) are still available for use. Should the 500 cfm flow of these fans be considered with the 3000 cfm of the 31-7213 or 31-7214 fans? This drawing indicates that the normal supply to the control rooms to be 12100 cfm and 7225 cfm. What portion of these flows is attributed to outside air makeup (prior to isolation)?

TVA Reply

The previous 500 cfm Emergency Pressurization Units have been designated as Auxiliary Pressurization Units. The power control leads for the fan motors have been lifted for these units and no credit is taken.

The 500 cfm flow of these units is not included in the 3000 cfm associated with the CREV filtration units.

The normal supply to the two control rooms is 11,600 cfm for Units 1 and 2 control room and 7225 cfm for the Unit 3 control room as indicated on FSAR figure 10.12-2a. The portion of these flows attributed to outside air make up is at least 900 cfm and 460 cfm respectively prior to isolation.

NRC Request B.5.

The LOCA analysis for control room dose assumes mixing and dilution in the base of the stack for a ground release for certain leakage paths. A review of the elevation and plan drawings of this area indicates the possibility of leakage plumes from the release point to the louvers in the stack walls affording little or no mixing. There does not appear to be sufficient internal structures in this area to justify the assumption of complete and timely mixing. Please provide a justification for this assumption.

TVA Reply

Only the lower floor of the stack is considered available for a mixing volume for "base of the stack" leakage. TVA Calculation ND-Q0065-920078, "Determine the Free Volume in the SGT Stack at Elevation 568 ft. to 597.5 ft.", determines the free volume in the bottom floor of the stack. For a cross-section of the base of the BFN plant stack see Figure 4 in the Enclosure to the August 10, 1994 letter (Reference 4). This is where any leakage past the back draft dampers would be expected to occur. With the exception of the Steam Packing Exhauster (SPE) back draft dampers, the expected leakage path would be through the back draft dampers and out into the room through the respective air dilution and cubicle exhaust fans. The lower floor of the stack is an enclosed area of approximately 76,800 ft^3, including a mezzanine area, with a free volume fraction of approximately 90 percent. The offgas dilution and cubicle exhaust fans are located in separate locations around the center of the room on the mezzanine area and are the expected entry path into the room for back draft damper leakage. The SPE ductwork is located on the opposite side of the room from the large louver to the outside. The mixing volume used in calculation ND-Q0031-920075 is approximately half (34560 ft^3) of the available free volume in the room.

Due to the following factors, substantial mixing will occur:

- With the exception of the SPE ductwork, the entry point for leakage into the room is in various locations around the approximate center of the room in the mezzanine area, above the large exit (louver) to the outside.

- The SPE ductwork is located on the opposite wall from the large exit (louver) to the outside.

- The room is an enclosed area.

- The exit (louver) from the room is located on the outer wall of the room.

- The leakage (10 SCFM) is drastically smaller than the free volume of the room.

TVA feels it is appropriate and conservative to use a mixing volume of approximately half of the room free volume.

C. QUESTIONS RELATING TO BOTH POWER UPRATE AMENDMENT REQUEST AND THE CREVS CORRECTIVE ACTIONS

NRC Request C.1.

FSAR text indicates that two SGTS trains can maintain the secondary containment at a negative pressure except for a short period at the beginning of a LOCA.

A. There are three SGTS trains, but only two emergency power trains. Please explain assumption that two SGTS trains would be available given a single failure of one emergency power train. Does the third SGTS train transfer automatically to the energized emergency bus, or is manual operator action necessary? Based on the emergency operating procedures, how long would it take for the second train to be energized? What is the differential pressure status of the secondary containment during this period.

TVA reply

Three 50 percent capacity SGTS trains auto start on a initiation signal. Each train has an independent power supply. They are each powered from a separate Emergency Diesel Generator and electrical board. There is no single failure that can affect multiple SGTS power supplies. Therefore, assuming a single failure, two 50 percent trains will be available for accident mitigation.

Since all three trains start initially and have independent power supplies, there is no automatic or manual transfer involved.

Two 50 percent capacity SGTS trains would remain energized, if one train failed. Therefore, differential pressure status of secondary containment is not affected.

B. Is the drawdown time for the secondary containment, with the various combinations of SGTS trains, measured periodically? If not what is the basis for FSAR conclusion that the draw down time is negligible? During periods of high exterior winds?

TVA Reply

Currently the integrity of the secondary containment is verified prior to each refueling outage by performing surveillance instruction, Combined Zone Secondary Containment Integrity Test. This test measures the amount of air being exhausted from the Reactor Building (RB) by the Standby Gas Treatment system, while maintaining the RB at -0.25 in. H_2O.

In anticipation of the implementation of Improved Standard Technical Specifications (ISTS), a new Surveillance Requirement (SR) is being developed that will include steps to verify the Reactor Building drawdown time. ISTS SR 3.6.4.1.3 requires that two trains of the Standby Gas Treatment System will reduce the RB pressure from ambient to -0.25 in. H_2O in less than two minutes. In preparation for these new requirements, during the performance of Combined Zone Secondary Containment Integrity Test in September of 1997, data was collected relating to the ability of SGTS to evacuate the RB from zero to -0.25 in. H_2O. Negative pressure recovered to -0.25 in. H_2O in approximately 35 seconds from the start of two SGTS trains. Adding an additional 40 seconds to account for diesel generator load sequencing that would occur in a worst case accident, a total drawdown time of 75 seconds is expected. These data confirm the conclusion in FSAR 14.6.3.6 regarding drawdown time.

Unfiltered release from the secondary containment building (ex-filtration) increases with high exterior wind during the drawdown period. However, the increased dispersion due the high wind would offset the effect of ex-filtration on dose. Additionally, this time period would occur at the beginning of the event before substantial amounts of fission products will have entered the reactor building from the primary containment.

NRC Request C.2.

The analysis appears to take credit for 90 percent filtration for both organic and non-organic iodine species. The BFN technical specification test acceptance criteria for methyl iodine is 10 percent penetration or bypass. This acceptance criteria does not support the assumed filtration credit. Standard technical specifications require the application of a safety factor to account for filter media degradation between tests. This factor is 5 for systems with heaters and 7 for systems without heaters. To assume 90 percent credit, the penetration or bypass fraction for filters with heaters must be less than 2 percent. Please provide a technical justification supporting the assumed filtration credit.

TVA reply

A review of the penetration data of both SGTS and CREV systems for the last ten years reveals data was within the 10 percent penetration limit of current TS. The worst case data point was 5.06 percent on SGTS train C. Because the filter media degradation between tests has not resulted in penetration greater than 10 percent, credit for 90 percent filtration is justified.

REFERENCES

1. TVA letter to NRC, dated May 5, 1992, Control Room Emergency Ventilation System (CREV)

2. TVA letter to NRC dated July 31, 1992, Resolution of Control Room Emergency Ventilation System (CREVS) Issues

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

4. TVA letter of NRC dated August 10, 1994, Response to Request For Additional Information Control Room Emergency Ventilation Systems (CREVS)

ENCLOSURE 2

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

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION 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)

COMMITMENT

TVA will provide a revised control room dose calculation, and its effect, if any, on the control room dose, in a supplement to this RAI response.