March 14, 2000 Template# NRR-058

Mr. J. A. Scalice
Chief Nuclear Officer and
Executive Vice President
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

SUBJECT: BROWN FERRY NUCLEAR PLANT, UNITS 2 AND 3 - ISSUANCE OF AMENDMENTS REGARDING LIMITS ON MAIN STEAM ISOLATION VALVE LEAKAGE (TAC NOS. MA6405 AND MA6406)

Dear Mr. Scalice:

The Commission has issued the enclosed Amendment Nos. 263 and 223 to Facility Operating Licenses Nos. DPR-52 and DPR-68 for the Browns Ferry Nuclear Plant, Units 2 and 3, respectively. These amendments revise the technical specifications limits for allowable main steam isolation valve leakage. They have been prepared in response to your application dated September 28, 1999 (TS399), as supplemented by letter dated February 4, 2000.

A copy of the Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

/RA/

William O. Long, Senior Project Manager, Section 2
Project Directorate II
Division of Licensing Project Management

Docket Nos. 50-260 and 50-296

Enclosures: 1. Amendment No. 263 to License No. DPR-52
2. Amendment No. 223 to License No. DPR-68
3. Safety Evaluation

cc w/enclosures: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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Executive Vice President
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William O. Long, Senior Project Manager, Section 2
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cc w/enclosures: See next page

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-260

BROWNS FERRY NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 263
License No. DPR-52

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated September 28, 1999, as supplemented by letter dated February 4, 2000, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-52 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 263, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Richard P. Correia

Richard P. Correia, Chief, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: March 14, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 263

FACILITY OPERATING LICENSE NO. DPR-52

DOCKET NO. 50-260

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
iii	iii
---	v
3.6-16	3.6-16
iii Bases	iii Bases
B 3.6-35	B 3.6-35
B 3.6-36	B 3.6-36

LIST OF FIGURES

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.5	Verify the isolation time of each power operated, automatic PCIV, except for MSIVs, is within limits.	In accordance with the Inservice Testing Program
SR 3.6.1.3.6	Verify the isolation time of each MSIV is ≥ 3 seconds and ≤ 5 seconds.	In accordance with the Inservice Testing Program
SR 3.6.1.3.7	Verify each automatic PCIV actuates to the isolation position on an actual or simulated isolation signal.	24 months
SR 3.6.1.3.8	Verify each reactor instrumentation line EFCV actuates to the isolation position on a simulated instrument line break signal.	24 months
SR 3.6.1.3.9	Remove and test the explosive squib from each shear isolation valve of the TIP System.	24 months on a STAGGERED TEST BASIS
SR 3.6.1.3.10	Verify leakage rate through each MSIV is ≤ 100 scfh and that the combined maximum pathway leakage rate for all four main steam lines is ≤ 150 scfh when tested at ≥ 25 psig.	In accordance with the Primary Containment Leakage Rate Testing Program
SR 3.6.1.3.11	Verify combined leakage through water tested lines that penetrate primary containment are within the limits specified in the Primary Containment Leakage Rate Testing Program.	In accordance with the Primary Containment Leakage Rate Testing Program

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.3.9

The TIP shear isolation valves are actuated by explosive charges. An in place functional test is not possible with this design. The explosive squib is removed and tested to provide assurance that the valves will actuate when required. The replacement charge for the explosive squib shall be from the same manufactured batch as the one fired or from another batch that has been certified by having one of the batch successfully fired. The Frequency of 24 months on a STAGGERED TEST BASIS is considered adequate given the administrative controls on replacement charges and the frequent checks of circuit continuity (SR 3.6.1.3.4).

SR 3.6.1.3.10

The analyses in References 1 and 5 are based on leakage that is less than the specified leakage rate. Leakage through each MSIV must be ≤ 100 scfh when tested at ≥ P_t (25 psig). The combined maximum pathway leakage rate for all four main steam lines must be ≤ 150 scfh when tested at ≥ 25 psig. If the leakage rate through an individual MSIV exceeds 100 scfh, the leakage rate shall be restored below the alarm limit value as specified in the Containment Leakage Rate Testing Program referenced in TS 5.5.12. This ensures that MSIV leakage is properly accounted for in determining the overall primary containment leakage rate. The Frequency is specified in the Primary Containment Leakage Rate Testing Program.

SR 3.6.1.3.11

Surveillance of water tested lines ensures that sufficient inventory will be available to provide a sealing function for at least 30 days at a pressure of 1.1 Pa. Sufficient inventory ensures there is no path for leakage of primary containment

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.11 (continued)

atmosphere to the environment following a DBA. Leakage from containment isolation valves that terminate below the suppression pool water level may be excluded from the total leakage provided a sufficient fluid inventory is available as described in 10 CFR 50, Appendix J, Option B.

Leakage through valves in closed loop seismic class I lines that are considered as extensions of primary containment present no potential for leakage to the environment. Leakage from these valves will be measured, but will be excluded when computing the total leakage. This leakage will be reported as required by the Primary Containment Leakage Rate Testing Program.

REFERENCES

1. FSAR, Section 14.6.
2. BFN Technical Instruction (TI), 0-TI-360.
3. 10 CFR 50, Appendix J, Option B.
4. FSAR, Section 5.2.
5. FSAR, Section 14.6.5.
6. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
7. FSAR Table 5.2-2.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-296

BROWNS FERRY NUCLEAR PLANT, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 223
License No. DPR-68

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated September 28, 1999, as supplemented by letter dated February 4, 2000, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-68 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 223, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Richard P. Correia

Richard P. Correia, Chief, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: March 14, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 223

FACILITY OPERATING LICENSE NO. DPR-68

DOCKET NO. 50-296

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
iv	iv
vii	vii
3.6-16	3.6-16
iv Bases	iv Bases
B 3.6-35	B 3.6-35
B 3.6-36	B 3.6-36

TABLE OF CONTENTS (continued)

(continued)

LIST OF FIGURES

SURVEILLANCE REQUIREMENTS (continued)

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TABLE OF CONTENTS (continued)

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.3.9

The TIP shear isolation valves are actuated by explosive charges. An in place functional test is not possible with this design. The explosive squib is removed and tested to provide assurance that the valves will actuate when required. The replacement charge for the explosive squib shall be from the same manufactured batch as the one fired or from another batch that has been certified by having one of the batch successfully fired. The Frequency of 24 months on a STAGGERED TEST BASIS is considered adequate given the administrative controls on replacement charges and the frequent checks of circuit continuity (SR 3.6.1.3.4).

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The analyses in References 1 and 5 are based on leakage that is less than the specified leakage rate. Leakage through each MSIV must be ≤ 100 scfh when tested at ≥ P_t (25 psig). The combined maximum pathway leakage rate for all four main steam lines must be ≤ 150 scfh when tested at ≥ 25 psig. If the leakage rate through an individual MSIV exceeds 100 scfh, the leakage rate shall be restored below the alarm limit value as specified in the Containment Leakage Rate Testing Program referenced in TS 5.5.12. This ensures that MSIV leakage is properly accounted for in determining the overall primary containment leakage rate. The Frequency is specified in the Primary Containment Leakage Rate Testing Program.

SR 3.6.1.3.11

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(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.11 (continued)

atmosphere to the environment following a DBA. Leakage from containment isolation valves that terminate below the suppression pool water level may be excluded from the total leakage provided a sufficient fluid inventory is available as described in 10 CFR 50, Appendix J, Option B.

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 263 TO FACILITY OPERATING LICENSE NO. DPR-52

AND AMENDMENT NO. 223 TO FACILITY OPERATING LICENSE NO. DPR-68

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNITS 2, AND 3

DOCKET NOS. 50-260, AND 50-296

1.0 INTRODUCTION

By application dated September 28, 1999, as supplemented by letter dated February 4, 2000, the Tennessee Valley Authority (TVA, the licensee) submitted an application to amend the Browns Ferry Units 2 and 3 (BFN-2/3) Technical Specifications (TS). The TS would be changed to increase the allowable leakage for any one of the four main steam line (MSL) penetrations from 11½ standard cubic feet per hour (scfh) to 100 scfh, and to establish a 150 scfh limit on the maximum allowable combined leakage of all four MSL penetrations. The 150 scfh limit is based on radiological dose considerations.

The February 4, 2000 letter was submitted in response to a staff request for additional information (RAI) dated November 23, 1999. In response to the RAI, the licensee provided additional information relating to the functionality and qualification of the MSL leakage path piping, and presented the results of new radiological dose analyses that replace those of the September 28, 1999 submittal which were based on extrapolation of the results of a previous analysis. The February 4, 2000 letter did not revise the application in a manner that would affect the conclusions of the initial determination of no significant hazards consideration or expand the scope of the staff's October 6, 1999 Federal Register notice.

The licensee's application also included a request for exemptions, for each unit, from Title 10, Code of Federal Regulations Part 50, Appendix J. The proposed exemptions would allow the licensee to exclude leakage from MSL penetrations from consideration in meeting the requirement that the total leakage of Type B and Type C containment penetrations shall not exceed 60% of the maximum allowable containment leakage (i.e., $0.6L_a$). MSL penetrations are Type C containment penetrations. This request is being reviewed concurrently as a separate licensing action.

2.0 BACKGROUND

The main steam system transports steam from the reactor vessel to the main turbines and other steam driven auxiliary equipment. Each of the four MSLs contains two quick closing main steam isolation valves (MSIVs) located in the containment penetration piping. One MSIV in each line is located inside the containment, and the other is located outside. These valves serve to rapidly isolate the primary containment MSL penetrations in the event of an MSL break accident or loss-of-coolant accident (LOCA). [Note: At some boiling water reactor (BWR) facilities, the MSLs are provided with a leakage control system (LCS) to collect and process MSIV leakage, however, the BFN facilities, do not have an MSIV LCS since such systems were not introduced until Standard Review Plan Section 6.7 was implemented.]

MSIVs, due to their size and service conditions, have a history of leakage in excess of their design criteria. (Specific information relating to the licensee's difficulties with MSIV leakage at BFN is provided in a licensee letter dated October 15, 1986 and in NRC Inspection Reports issued on March 19, 1987.) On July 16, 1982, the staff issued Information Notice 82-23, "Main Steam Isolation Valve Leakage," which discussed the high frequency at which MSIVs were failing to meet TS leak test criteria. Because of these recurring problems with excessive leakage of MSIVs, Generic Issue C-8, "Main Steam Line Valve Leakage Control Systems," was established. The same year, the BWR Owners' Group (BWROG) formed an MSIV Leakage Committee to address the MSIV leakage issue. In 1986, Generic Letter 86-17, "Technical Findings Related to Generic Issue C-8; Boiling Water Reactor Main Steam Isolation Valve Leakage and Leakage Treatment Methods," was issued and a follow-on MSIV Leakage Closure Committee was formed to further the effort. Based on the committee's work, the BWROG developed an approach for resolution of Generic Issue C-8 that proposed to remove the safety-related leakage control systems on those facilities having them, and increase MSIV allowable leakage limits. The BWROG described the proposal in "BWROG Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems," NEDC-31858P, Revision 1, dated October 1991.

The purpose of NEDC-31858P was to define a means by which BWR licensees could demonstrate to regulators that alternate leakage treatment (ALT) leakage pathways using main steam system piping and the main condenser are capable of performing a post-accident dose mitigation function for MSIV leakage, under safe-shutdown earthquake conditions. This would provide a basis to (1) eliminate MSIV Leakage Control Systems at those facilities having them and (2) increase allowable MSIV leakage rates. The staff reviewed NEDC-31858P and issued its safety evaluation on March 3, 1999 (Assession No. 9903110303), approving it for reference in future individual plant applications. The safety evaluation concluded that licensees demonstrating certain plant-specific attributes are eligible for amendments increasing the allowable MSIV leakage rate up to 200 scfh. Licensees would have to demonstrate that the main steam piping from the outermost isolation valve up to the turbine stop valve, the bypass/drain piping to the main condenser, and the main condenser, will retain their structural integrity during and following a safe-shutdown earthquake (SSE). NEDC-31858P has been reissued as NEDC-31858P-A, forwarded by letter from W. Glenn Warren to USNRC Document Control Center dated November 22, 1999.

3.0 <u>DISCUSSION AND EVALUATION</u>

3.1 <u>Scope of Staff Review</u>

The staff's review of the BFN amendment application encompassed three technical areas: (1) the functional design of the ALT path and capability to establish the ALT path under post-accident conditions, (2) structural/seismic issues related to integrity of the ALT path, and (3) the radiological consequences analyses of increased MSIV leakage via the ALT path.

3.2 Functional Design and Reliability of the ALT Boundary

The BFN ALT system utilizes the main steam lines and main steam line drain lines to direct MSIV leakage to the main condenser. This ALT path takes advantage of the capability of the large volume of the MSLs and condenser to hold-up and plate-out fission products in the MSIV leakage effluent. To mitigate a design-basis accident (DBA), this path must be available under DBA conditions with loss of offsite power (LOSP).

The ALT path is from the downstream side of the MSIVs through four 3-inch lines which join a 4-inch drain header to the main condenser. In addition to the MSL drains, the drain header also receives drains from high-pressure coolant injection and reactor core isolation coolant steam lines and auxiliary boiler. All valves in the flow path are normally open, with the exception of two, FCV-1-58 and FCV-1-59, which are normally closed. FCV-1-59 has a 4-inch bypass containing no valves or orifices and is, thus, of no concern with respect to ALT path availability.

In the event of an accident, operator actions will establish the primary ALT path to the main condenser. Normally-closed valves FCV-1-58 and FCV-1-59 will be opened using hand switches located in the main control room. These valves are powered by essential busses with emergency diesel generator backup and will be included in the Inservice Testing Program. In Unit 2, FCV-1-58 is powered by 480 V Reactor Motor Operated Valve (RMOV) Board 2C. RMOV Board 2C is normally aligned to 480 V Shutdown Board 2B which is Division II power. An alternate power source to RMOV Board 2C is provided by 480 V Shutdown Board 2A which is Division 1 power. If the normal feeder (480 V Shutdown Board 2B) to RMOV Board 2C is lost, it can be transferred to its alternate power supply (480 V Shutdown Board 2A) by remote breaker operation. The two 480 V Division I and II Shutdown Boards both have their own (separate) diesel generator power supplies. Therefore, it is a simple operation to transfer 480 RMOV Board C to its alternate power supply. The arrangement for Unit 3 is similar. These arrangements provide a high degree of electrical power reliability for FCV-1-58.

The licensee considered the action of including FCV-1-58 in the Generic Letter 89-10 motor operated valve augmented testing program but determined that the American Society of Mechanical Engineers Code testing requirements are adequate due to the fact this valve is not subject to high dynamic loads under the proposed accident conditions.

A sealing steam supply valve, PCV 1-147, will be modified so that it fails closed instead of open, and check valves will be added to preheater steam lines to ensure ALT boundary integrity. Subsequently, all of the valves in the ALT boundary will be either: (a) normally closed manual valves, (b) normally closed motor-operated valves, (c) fail-close air-operated valves, or (d) check valves with spring-assisted closure. The licensee determined that potential loss of sealing steam due to the PCV 1-147 modification would have no adverse effects on postulated events. The new check valves will be periodically tested under the scope of the Inservice Testing Program.

Section 5.2 of the March 3, 1999 safety evaluation states that a secondary path to the condenser, having an orifice, should exist. NEDC-31858P does not require that this secondary path have the same flow capability as the primary path. The licensee's application states that in the event FCV-1-58 were to fail to open, the leakage flow would split, with part of the flow going to the condenser via a 0.1875-inch-diameter orifice in a normally open bypass around FCV-1-58, and the remainder going to the condenser via normal leakage paths through the main steam stop/control valves and through the high pressure turbine. The functional design of the secondary path is consistent with the criteria of NEDC-31858P and is acceptable.

The licensee has committed to establish training and procedures for line-up of the primary ALT path in the event of an accident.

The staff has reviewed the proposed primary ALT path and concludes that it meets the NEDC-31858P functional design and reliability criteria for an "isolated condenser" ALT path and would be available under post-accident conditions including LOSP.

3.3 Structural Integrity of the ALT Pathway

Systems and equipment needed to mitigate design basis accidents are required to be designed and constructed to Seismic Category I criteria in order to withstand earthquakes. Because the portions of the BFN main steam piping, drain lines, and main condenser that are located outside of the primary containment were not specifically designed to Seismic Category I criteria, the licensee utilized other means of demonstrating that the ALT system is seismically rugged and meets General Design Criterion (GDC) 2 of Appendix A to 10 CFR Part 50. The methods and criteria used were developed by the BWROG and are described in NEDC-31858, Revision 2, which, as noted above, serves as the generic basis for acceptability of individual licensee applications. The staff reviewed the BWROG report and found that its approach of utilizing earthquake experience data to demonstrate the seismic ruggedness of nonseismically-designed main steam piping and condensers, as supplemented by plant-specific seismic verification walkdowns and analytical evaluation, to be an acceptable means of demonstrating that the ALT system is seismically rugged and meets GDC 2 of Appendix A to 10 CFR Part 50.

In support of NEDC-31858, the BWROG retained Earthquake Engineering, Inc. (EQE) as a consultant to conduct a review of earthquake experience data on the performance of main steam piping systems and equipment, including large condensers in non-nuclear facilities that have experienced strong earthquake motions. EQE also compared the design practices utilized in construction of these piping systems, to those typically used in domestic BWRs. The result of the comparison supported the BWROG position that BWR main steam piping and condensers would maintain their pressure-retention capability during a design basis earthquake. TVA's application for BFN utilizes the NEDC-31858 methodology.

The licensee's application included a report identified as "Browns Ferry Nuclear Plant Increased MSIV Leakage Tech Spec Change Submittal - Seismic Evaluation Report," 200918-R-002, Revision 0, August 31, 1999, prepared for TVA by EQE International. This report summarizes the engineering activities performed for the main steam piping seismic verification to support the proposed amendment. The seismic verification activities were performed in accordance with the NEDC-31858 recommendations, and consisted of the following key elements:

- Seismic Experience Database Comparisons
- Seismic Verification Walkdowns
- Seismic Assessments of Selected Components

Seismic Experience Database Comparisons: The main steam piping and condensers in the (mostly fossil fueled) earthquake experience database exhibited substantial seismic ruggedness even though not typically designed to resist earthquakes. By showing that the BFN main steam system and condenser are of comparable construction and that BFN's plant-specific ground response spectrum is bounded by those of the earthquake experience database, it is reasonable to assume that BFN's equipment would survive an SSE earthquake.

To establish the applicability of NEDC-13858P, Revision 2 regarding usage of the earthquake experience-based methodology for demonstrating seismic ruggedness, comparisons of the BFN ground response spectra were made with those of selected database facilities chosen from those approved in the staff's March 3, 1999 safety evaluation. TVA selected 10 of the 13 for comparison to BFN. Based on the comparison, the BFN ground spectrum is generally bounded by the earthquake experience database sites at the frequencies of interest.

The BFN ALT piping is of welded steel with standard support components. The system generally meets U.S. of America Standards (USAS) B31.1-1967 criteria. Overall the BFN ALT piping is similar to that of the database facilities.

Based on the piping design and response spectrum comparisons, the use of the NEDC-31858 earthquake experience-based approach is applicable to BFN.

Seismic Verification Walkdowns and Assessments: Section 5.5 of the staff's March 3, 1999, safety evaluation states that walkdowns will be performed by the licensee to identify outlier design features that could constitute potential failure modes. Guidelines for walkdown teams are described in NEDC-31858. Outliers typically fall into four categories: (1) failure of pipe due to excessive displacements of attached equipment, (2) failure of branch piping due to excessive displacement of attached piping mains, (3) failure of piping associated with loss of pipe supports, and (4) failure of piping due to failure of enclosing building and its internals, including collapse of masonry walls and interactions between piping and nearby components.

Specific ALT path piping/component attributes that are screened in walkdowns include:

- Piping with dead weight support spacing in excess of the B31.1 suggested spans, and tubing with excessive sagging
- Heavy, unsupported components
- Non-ductile materials such as cast iron or PVC
- Non-standard fittings or unusual attachments that could cause excessive localized stresses
- Pipe supports that exhibit non-ductile behavior
- Presence of severe corrosion.

- Areas of potential seismic interactions of ALT path components and other systems and equipment

The results of the BFN-2/3 walkdowns are described in the EQE Report. The walkdown boundaries encompass the entire ALT path as described above.

The walkdowns were performed by degreed engineers having 10 to 20 years experience in structural engineering and/or earthquake engineering application to nuclear power plants.

Some of the outlier conditions identified by the walkdowns included:

- short rod hangers
- support spans exceeding USAS recommendations
- differential displacements of main pipe and branch lines
- equipment anchorage deficiencies
- valve performance evaluation
- condenser and condenser anchorage evaluation
- proximity and potential impact of piping with equipment, structural features and other piping, i.e., seismic interactions.

For BFN-3, 34 conditions which did not conform to guidelines were found. All BFN-3 outliers will be fixed during the forthcoming (Spring 2000) outage. For BFN-2, 26 outliers were found, and more may be found as a result of additional walkdowns to be performed to expand the walkdown scope to equal that of BFN-3, and thereby encompass the entire ALT path. All BFN-2 outliers will be resolved prior to startup from the Cycle 11 outage in Spring 2001.

Seismic Assessments of Selected Components:

Seismic Demand: For seismic evaluations and outlier resolution, the horizontal seismic demand for components located within about 40 feet of the turbine building grade elevation was taken as the 5% damped design basis event input spectrum (0.2g Housner) scaled by 1.6 to account for soil amplification, consistent with BFN's design basis seismic criteria for soil founded structures, and 1.5 for building amplification, consistent with the Seismic Qualification Users Group Generic Implementation Procedure. For components located above 40 feet of building grade, an additional amplification factor of 1.5 was applied. For the vertical direction, seismic demand was taken as 2/3 that of the horizontal direction, with a soil amplification factor of 1.1 instead of 1.6.

Seismic capacity: Seismic assessments and outlier resolutions utilized the following load combinations and allowable stresses:

Component	Load Combination	Stress Allowables
Piping	D + P + I + A (Primary + Secondary)	2.0 Sy
Pipe Supports	D + T + I + A	AISC
Equipment Anchorages	D + I	AISC, GIP
Valve	3g Load Check	GIP

where,
D = Dead Load
P = Pressure Load
T = Thermal Load
I = Seismic Inertial Load
A = Load Due to Anchor Movement
Sy = Material Yield Strength at Temperature
AISC = American Institute of Steel Construction
GIP = Generic Implementation Procedure

Piping and Supports: The Browns Ferry ALT system is constructed of welded steel pipe and standard support components. Support spacing is generally consistent with USAS B31.1 recommendations. Pipe diameters vary from 1.315 inches to 24 inches. Bounding evaluations were performed for typical support configurations using the above demand/capacity criteria. Stress ratios for anchor bolts do not exceed 0.73, and for overhead weld attachments, do not exceed 0.70.

Turbine Building: The ALT path lies within the lower portion of the turbine building which is classified as a seismic Class II structure, i.e., not essential for the mitigation of the consequences of design basis accidents. Performance of the turbine building structures during a seismic event is of interest only to the extent that the building structure and internal components not adversely affect the ALT pathway and condenser.

The turbine building is a reinforced concrete structure below the operating floor (elevation 617 feet), and is supported on steel H-piles to bedrock. The superstructure is framed by transverse welded steel rigid frames which expand approximately 107 feet. Frames are designed with fixed bases to resist lateral forces from overhead cranes with wind loads to 100 mph. An expansion joint is provided between a two-bay frame for Units 1 and 2, and a single bay frame for Unit 3. The design of the steel superstructure is based on AISC "Manual of Steel Construction," 6th Edition.

The compressive strength of the structural concrete (f'_c) is 3000 pounds per square inch (psi) at 28-days cure time, except that turbine building columns are 4000 psi. For evaluation/reanalysis of the structure, long-term concrete strength gain may be used. Reinforcing steel used is in accordance with American Society for Testing and Materials (ASTM) A432 Grade 60 or ASTM A615 with f_y = 60,000 psi and E_s = 29 x 10^6 psi. Beams and slabs have been designed by American Concrete Institute (ACI) working stress methods and columns designed by working stress method, and checked by ACI ultimate strength design method using a load factor of 1.8.

Unreinforced masonry walls are particularly vulnerable to seismic damage. Where masonry walls exist in the turbine building, they are generally used as removable shield walls or non-load bearing partition walls. Since non-reinforced masonry walls do not perform well during seismic events, masonry walls were specifically reviewed during the seismic verification walkdowns

During the walkdowns, particular attention was given to interactions with the turbine building masonry walls. Outliers were identified and resolved.

Based on the design bases of the BFN turbine building, and the seismic performance of similar types of industrial structures in past strong-motion earthquakes, the BFN turbine building is expected to remain intact following a design basis earthquake.

Condensers: The BFN condensers consist of three single-pass, single pressure, radial flow type surface condensers. Each condenser is located beneath each of the three low pressure turbines, and is structurally independent. Table 4-8 of the EQE Report lists the design data for the BFN condensers and an earthquake experience database site listed in the NEDC-31858P, Revision 2. In addition, design characteristic comparisons of the BFN condensers with the selected database condensers are shown in Figures 4-2 through 4-5 of the EQE Report. The BFN condenser design data is comparable to the data for the database site. The BFN condensers were also analytically evaluated for structural integrity subject to seismic design-basis earthquake (DBE) loads. The condenser was conservatively modeled as a 50'Lx47.5'Hx32'W four-walled tank containing sufficient water, at the proper elevation, to provide the same center of gravity and weight as an actual condenser at operating condition. This model was analyzed for a 5% damped ground spectrum with a 0.32 g horizontal acceleration and 0.2 g (2/3 horizontal) vertical acceleration. The four walls were assumed to be the same material as the condenser shell, i.e., 7/8 inch thick steel. Maximum stress ratios, based on AISC allowables, are 0.12 for combined axial and bending and 0.10 for shear.

The condenser support anchorage consists of a center key and six support feet. The center support is a fixed anchor and consists of a built-up wide flange H section embedded 4 feet into the concrete pedestal, which is connected to the turbine building base mat and welded to the bottom plate of the condenser. The support plates consist of two to three anchors of 2- to 2-1/2-inch diameter bolts. Each anchor bolt has greater than 5-feet nominal length with approximately 48 inches of embedment into the concrete pedestal, which is connected to the turbine building base mat. These supports are designed to resist vertical operating loads and are slotted radially from the center key to allow for thermal growth. Shear forces are transferred to the wide flange-shaped anchor in the center and to the anchor bolts and shear keys to the support feet and carried through the concrete pedestal to the turbine building base mat.

The anchorage for the BFN condenser is comparable with the performance of the anchorages for similar condensers in the earthquake experience database. The shear areas of the condenser anchorage, in the directions parallel and transverse to the turbine generator axis, divided by the seismic demand, were compared with those presented in NEDC-31858P, Revision 2. The BFN condenser anchorage shear area to seismic demand is substantially greater than the selected database sites. The condenser support anchorage was also analytically evaluated and the results indicate that the combined seismic DBE and operational demand are less than the anchorage capacity based on the AISC allowables. Maximum stress ratios are 0.70 for bolt tension in the perimeter support feet, and 0.86 for shear in the center support built-up section.

The licensee's application includes a table "Comparison of Browns Ferry and Selected Database Condensers," which compares the BFN condenser design with that of Moss Landing Units 6 and 7 experience database facility. The Moss Landing facility experienced a strong seismic motion (est. 0.34 free field peak ground acceleration) during the 1989 Loma Prieta earthquake. The seismic accelerations for that facility bound BFN's for all frequencies of interest. The data in the table indicate that the BFN condensers can be expected to exhibit equal or better seismic performance. A detailed discussion of this earthquake and its effects on the Moss Landing facilities is provided in NEDC-31858, Revision 2, Appendix D, Section 3.3.1. The Moss Landing Units 6 and 7 condensers are representative of those found in nuclear facilities. Moss Landing experienced greater than 0.2 g horizontal acceleration, but no condenser damage was experienced.

The above comparisons of the condenser seismic experience data and the anchorage capacity evaluations demonstrate that the conclusions presented in the NEDC-31858P, Revision 2, can be applied to the BFN condensers. That is, a significant failure of the condenser in the event of a DBE at BFN is highly unlikely and contrary to the large body of historical earthquake experience data.

Piping Supports and Anchorages: Section 5.7 of the March 3, 1999, safety evaluation states that representative supports and anchors associated with ALT piping should be analytically evaluated for seismic adequacy. The licensee reviewed the design codes and standards, piping design parameters and support configurations. The BFN piping was found to be similar to and well-represented by that of the experience database facilities. Bounding evaluations were performed for typical support configurations. The evaluations provide reasonable assurance that the ALT piping, related supports, and components will remain functional in the event of a DBE.

Conclusion: The staff has reviewed the licensee's seismic experience database comparisons, and the results of the seismic verification walkdowns and equipment seismic assessments. Based on the information presented by the licensee, the staff concludes that the ALT pathway has sufficient structural integrity requirement that it would be functional under design basis accident conditions.

3.4 Radiological Consequences of Design Basis Accidents

3.4.1 Background

The staff reviewed the licensee's revised analyses of radiological consequences. The purpose of this portion of the review was to (1) determine if the licensee utilized acceptable analytical methodology, and (2) determine if the calculated doses are within the limits of regulatory acceptance criteria.

Two previous licensing actions were found to have bearing on the current amendment application. By letter dated July 31, 1992, TVA described corrective actions that were to be carried out to resolve deficiencies in the control room emergency ventilation system at BFN-1/2/3 Unit Nos 1, 2 and 3. In another letter dated October 1, 1997, TVA submitted a license amendment request for a power uprate. In its review of the power uprate application, the staff questioned TVA's technical basis for treating the MSIV leakage as a separate pathway in control room assessments while treating the leakage as part of the filtered containment leakage pathway in offsite dose assessments. The staff asked TVA to reconsider this analysis approach. In addition, the staff noted that TVA had not satisfied the prerequisites identified in the BWROG report. TVA agreed to re-assess the radiological consequences of a design basis LOCA to the control room and to persons located offsite and to make necessary modifications to satisfy prerequisites. TVA proposed a license condition (imposed by letter dated September 8, 1998) stipulating this commitment. By letter dated March 30, 1999, TVA submitted the required analysis and the NRC accepted this analysis by letter dated August 3, 1999. That action incorporated the MSIV leakage component and the BWROG analytical methodology into the BFN design basis.

In its September 28, 1999 application, TVA proposed to increase the allowable MSIV leakage rate from the current 11.5 scfh per valve to 100 scfh per valve, with the additional limitation that the combined leakage from all four main steam lines cannot exceed 400 scfh. To support this increase, TVA re-analyzed the radiological consequences of the design basis LOCA to the control room and to persons offsite, using the BWROG methodology used in the earlier analyses, but scaled to reflect the increased MSIV leakage. In response to questions raised by the staff in a subsequent request for additional information, and discussed below, TVA further re-analyzed the radiological consequences of the proposed increase in allowable leakage. As a result of this reanalysis, TVA modified their application by letter dated February 4, 2000, proposing a technical specification (TS) limit of 100 scfh per valve provided the combined leakage from all four MSLs is not greater than 150 scfh.

3.4.2 Release Pathways

Limitations on primary containment leakage rates provide assurance that the total containment leakage (L_a) will not exceed the value assumed in the design basis LOCA analysis described in Chapter 14 of the FSAR. The radiological analyses consider the effect of containment leakage on both the offsite doses and the control room doses. In the BFN plant configuration, leakage from the containment is contained within the reactor building (secondary containment), collected and filtered through the standby gas treatment system (SBGT) and discharged to the environment via the plant stack. MSIV leakage provides a path for radioactivity in the containment to be released to the main steam lines and to bypass the secondary containment. As a result, MSIV leakage would not be collected and processed through the SBGT, but would enter the environment as an unfiltered ground level release, a less conservative situation.

TVA analyzed the radiological consequences of the MSIV leakage separately from the containment leakage and summed the contribution of each pathway (along with the postulated doses for the emergency core cooling system (ECCS) leakage pathway) to obtain the total doses for comparison to the 10 CFR Part 100 dose guidelines and the General Design Criteria (GDC)-19 control room criteria. TVA analyzed the MSIV leakage pathway using a leakage value greater (i.e., 168 scfh) than the proposed TS allowable value of 150 scfh, and analyzed the containment leakage pathway using the 2% vol./day TS L_a value. This is conservative.

In performing the power uprate dose consequences analyses, TVA used the approved fission product transport methodology of the BWROG topical report NEDC-31858P, Revision 2. For this license amendment request, TVA scaled the results obtained in the power uprate analyses to reflect changes in χ/Qs, as described above, and the increased leakage. The staff found this approach to be unacceptable because fission product attenuation in the steam lines and the main condenser is not directly proportional to the steam flow associated with the MSIV leakage and the linear scaling approach would therefore underestimated the postulated doses. In response to this concern, TVA re-analyzed the doses. As a result of the reanalysis, TVA reduced the leakage assumption to a combined leakage for all steam lines of 168 scfh, a value greater than the proposed TS limit for combined leakage of 150 scfh. The staff performed independent calculations and confirmed that the licensee's revised analyses produce conservative results. Parameters used by the staff in its independent analysis are identified in Table 1 (Attachment 1).

Besides analyzing the dose consequences of the increased MSIV leakage, TVA also re-analyzed the containment and ECCS leakage pathways. These reanalyses used inputs, assumptions, and methodologies previously approved by the staff along with the proposed revised χ/Q values. The postulated consequences of these two release pathways were summed with the postulated consequences of the MSIV leakage pathway to obtain the total offsite and control room LOCA doses.

3.4.3 Meteorology

For the new radiological consequences analyses, TVA re-evaluated certain χ/Q values used in the calculations. The χ/Q values for locations offsite were comparable to those used in previous approved analyses and were retained for the new analyses. However, the control room χ/Q valves were revised using the ARCON96 computer code described in NUREG/CR-6331. This analysis methodology is an acceptable alternative for the dispersion model in the control room habitability assessment procedure of Murphy and Campe. Building wake diffusion model studies conducted for the NRC showed that the Murphy-Campe method does not predict the χ/Q values in the wake of buildings accurately and may overpredict these χ/Q values during low wind speed conditions. Because of the use the ARCON96 code with its more mechanistic algorithm, many BFN control room χ/Q values (i.e., the ground level values) decreased in magnitude. For sites with control room ventilation intakes that are close to the base of tall stacks, such as at BFN, ARCON96 may underpredict the χ/Q values for stack releases. In response to this phenomenon, TVA re-evaluated the χ/Q values for the stack using the methods of Regulatory Guide 1.145 and 1.111, and used these revised values in re-assessing the consequences of the increased MSIV leakage for the stack release cases. Table 2 (Attachment 2) tabulates the acceptable χ/Q values used by TVA in the analyses supporting the increased MSIV leakage.

3.4.4 Control Room Habitability

The staff also reviewed the licensee's revised control room operator dose analyses. The staff is currently working toward resolution of generic issues related to control room habitability, in particular, the validity of control room infiltration rates assumed by licensees in analyses of control room habitability. Recent tests conducted at 20% of operating reactors have shown that, in all cases, the measured infiltration rates exceeded the values assumed in the design basis analyses. While in each case the affected licensees were able either to reduce the excessive infiltration or show the acceptability of the observed infiltration, the collective experience has caused concerns regarding those facilities that have not performed the enhanced testing. The staff is currently participating in an NRC-industry initiative to resolve these concerns.

By letter dated July 31, 1992, TVA described several corrective actions to resolve deficiencies in the design of the control room emergency ventilation system at BFN Unit Nos. 1, 2 and 3. These actions were self-initiated by TVA on discovery that, contrary to the design basis, there was substantial unfiltered infiltration to the BFN control room. These actions included installation of new pressurization fans, relocation of ventilation intakes, and other actions designed to reduce the unfiltered inleakage.

The staff has determined that there is reasonable assurance that the BFN control room will be habitable during design basis accidents and that this amendment may be approved before the resolution of this generic issue. The staff bases this determination on (1) the relative magnitude of the infiltration currently assumed in the BFN analyses (3717 cfm unfiltered infiltration), (2) the favorable site χ/Q values, (3) the testing and other actions already taken by TVA, (3) the availability of potassium iodide (KI) as an interim compensatory measure, and (4) the low probability of design basis events, occurring during this interim period, that could result in radioactivity releases sufficient to challenge the ability of control room personnel to protect the health and safety of the public. The approval of this amendment and exemption does not exempt TVA from regulatory actions that may be imposed in the future as this generic issue is resolved.

3.4.5 Existing License Condition

In the power uprate amendments license conditions were imposed on TVA. The following condition is relevant to the current action:

> TVA will maintain the ability to monitor radiological conditions during emergencies and administer potassium iodine to the control room operators to maintain doses within GDC-19 guidelines. This ability will be maintained until the required modifications, if any, are complete.

The required modifications identified in this condition are scheduled to be implemented during the planned Unit 2 outage in the spring of 2001. While this condition was originally imposed as a compensatory measure related to MSIV leakage, the availability of KI compensates for potential uncertainty in the measured infiltration rate during the interim period in which resolutions are being developed for the generic control room habitability issues.

3.4.6 Conclusion Regarding Radiological Consequences Analyses

The staff's radiological review encompassed the following areas: (1) release pathways, (2) meteorology and (3) control room habitability, and focused on the changes from the previously accepted analyses. Based on its review of the information provided by the licensee, the staff has concluded that the offsite dose consequences of design basis accidents are within the acceptance criteria of 10 CFR Part 100, and the control room dose consequences are within the acceptance criteria of GDC-19.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Alabama State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32 and 51.35, an environmental assessment and finding of no significant impact was published in the Federal Register on February 29, 2000 (65 FR 10844). Accordingly, based upon the environmental assessment, the Commission has determined that issuance of this amendment will not have a significant effect on the quality of the human environment.

4.0 CONCLUSION

Based on the information provided by TVA related to increasing the allowable MSIV leakage rate to 100 scfh per valve provided the combined leakage from all four MSLs is not greater than 150 scfh, the staff finds reasonable assurance that the radiological consequences of the design basis LOCA accident at BFN-2/3 will be less than the dose guidelines of 10 CFR Part 100 and the criteria of 10 CFR Part 50, Appendix A, GDC-19 and Section 6.4 of NUREG-0800.

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Attachments: 1. Table 1
2. Table 2

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Date: March 14, 2000

Table 1

BFN Accident Analysis Parameters Used by Staff

Reactor power (3458 x 1.02), MWt	3527
Iodine species distribution	
Elemental	0.91
Organic	0.04
Particulate	0.05
Main condenser volume, ft^3	125,000
SGTS Flow, cfm	
Stack, Elevated	24,750
Damper bypass, ground level	10
SGTS drawdown time, sec *(assume release is ground level during drawdown)*	75
SGTS Filter Efficiency, all species, %	90
Dose conversion factors	FGR11/FGR12
Breathing rate, offsite, m^3/s	
0-8 hours	3.47E-4
8-24 hours	1.75E-4
>24 hours	2.32E-4
Breathing rate, control room, m^3/s	3.47E-4
Control room unfiltered infiltration, cfm	3717
Control room filtered pressurization, cfm	3000
Control room volume, ft^3	210,000
Control room intake filter efficiency, all species, %	90
Control room occupancy factor	
0-24 hrs	1.0
1-4 days	0.6
4-30 days	0.4
Containment Leakage Source	
Core release fraction to containment	
Iodine	0.25
Noble gases	1.0
Primary containment volume, ft^3	283,000
Containment leak rate, %/day	2.0
Secondary containment volume (50% of free volume)	1,932,000
SGTS ground level leakage (base of stack), cfm	10

Volume at base of stack (50% of free volume), ft^3	34,560
X/Q	Table 1, Top & Base of Stack
<u>AD System Release</u>	
Activity same as containment leakage case	
Flow rate, cfm	139
CAD operation, days post accident	10, 20, 29
CAD operation duration, hours	24
No mixing in reactor building	
X/Q	Table 1, Top & Base of Stack
<u>MSIV Leakage</u>*	
Activity same as containment leakage case	
MSIV leak rate, ft^3/hr	168
Release from main condenser, scfm	1.77
Plateout fraction	0.9
X/Q values	Table 1, Turbine Building
<u>ECCS Leakage</u>	
Core release fraction to containment sump	
Iodine	0.5
Noble gases	0.0
Suppression pool liquid volume, ft^3	141,260
Estimated leakage, gpm	5
Iodine Flash Fraction	0.1
Release mixes in secondary containment and released via SGTS	
X/Q	Table1, Top & Base of Stack

**Staff values used to confirm BWROG methodology result.*

Table 2

BFN METEOROLOGY

Time Period	Control Room* Unit 1	Control Room* Unit 3	Site Boundary EAB	LPZ
Top of Stack Releases				
0-0.5 hrs	3.40E-5	3.02E-5	2.40E-5	1.26E-5
0.5-2 hrs	9.08E-13	1.41E-7	9.70E-7	1.13E-6
2-8 hrs	3.41E-13	4.50E-8		5.75E-7
8-24 hrs	2.09E-13	2.54E-8		4.10E-7
1-4 days	7.21E-14	7.36E-9		1.97E-7
4-30 days	1.57E-14	1.24E-9		6.88E-8
Base of Stack Releases				
0-2 hrs	2.00E-4	8.60E-5	2.62E-4	1.31E-4
2-8 hrs	1.28E-4	6.46E-5		6.61E-5
8-24 hrs	5.72E-5	2.80E-5		4.69E-5
1-4 days	4.05E-5	2.00E-5		2.33E-5
4-30 days	3.09E-5	1.53E-5		7.96E-6
Turbine Building Releases				
0-2 hrs	1.20E-4	2.17E-4	2.62E-4	1.31E-4
2-8 hrs	9.96E-5	1.64E-4		6.61E-5
8-24 hrs	4.85E-5	7.89E-5		4.69E-5
1-4 days	3.15E-5	4.33E-5		2.23E-5
4-30 days	2.02E-5	3.35E-5		7.96E-6

* Pursuant to NUREG-0800 page 6.4-10, the χ/Q values for dual inlet designs without manual or automatic selection control may be based on the least favorable intake location with the value reduced by a factor of 2 to account for dilution effects. The control room χ/Q values are divided by two prior to use in dose analyses.

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