November 8, 2004

Mr. Karl W. Singer
Chief Nuclear Officer and
Executive Vice President
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6A Lookout Place
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Chattanooga, TN 37402-2801

SUBJECT: BROWNS FERRY NUCLEAR PLANT, UNITS 1, 2, AND 3 — CORRECTIONS TO AMENDMENT CONCERNING ALTERNATIVE SOURCE TERM (TAC NOS. MB5733, MB5734, AND MB5735) (TS-405)

Dear Mr. Singer:

On September 27, 2004, Amendment Nos. 251, 290, and 249 to Facility Operating License Nos. DPR-33, DPR-52, and DPR-68 for the Browns Ferry Nuclear Plant (BFN), Units 1, 2, and 3, respectively were issued. These amendments were in response to your application dated July 31, 2002, as supplemented by letters dated December 9, 2002, February 12, March 26, July 11, and July 17, 2003, and May 17, July 2, August 24, and September 17, 2004. These amendments adopted the alternative source term methodology by revising the current accident source term and replacing it with an accident source term as prescribed in Title 10 to the *Code of Federal Regulations* Section 50.67. The amendments also revised the Technical Specification sections associated with control room emergency ventilation, standby gas treatment, standby liquid control, and secondary containment systems.

Your BFN Licensing staff noted omissions of license pages; page 5 for Unit 2 and page 4 for Unit 3. In addition minor editorial errors were noted in the associated safety evaluation (SE). These editorial changes have been made in the pages as indicated by margin bars. Formatting changes in the Tables are not indicated by margin bars.

These changes do not affect the conclusion of the SE. We apologize for any inconvenience this may have caused.

Sincerely,

/RA/

Eva A. Brown, Project Manager, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-259, 50-260 and 50-296

Enclosure: Corrected Pages

cc w/encls: See next page

Mr. Karl W. Singer
Tennessee Valley Authority

BROWNS FERRY NUCLEAR PLANT

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Docket Nos. 50-259, 50-260 and 50-296

Enclosure: Corrected Pages

cc w/encls: See next page

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ATTACHMENT TO LICENSE AMENDMENT NO. 290

FACILITY OPERATING LICENSE NO. DPR-52

DOCKET NO. 50-260

Replace pages 4 and 6 of Operating License No. DPR-52 with the attached pages.

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
3.1-23	3.1-23
3.1-24	3.1-24
3.1-25	3.1-25
3.1-26	3.1-26
3.3-65	3.3-65
3.3-70	3.3-70
3.6-44	3.6-44
3.6-45	3.6-45
3.6-47	3.6-47
3.6-49	3.6-49
3.6-51	3.6-51
3.6-52	3.6-52
3.6-53	3.6-53
3.7-9	3.7-9
3.7-10	3.7-10
3.7-11	3.7-11
- - -	3.9-22
- - -	3.9-23
- - -	B3.9-36
- - -	B3.9-37
- - -	B3.9-38

ATTACHMENT TO LICENSE AMENDMENT NO. 249

FACILITY OPERATING LICENSE NO. DPR-68

DOCKET NO. 50-296

Replace pages 4 and 5 of Operating License No. DPR-68 with the attached pages.

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

Remove	Insert
3.1-23	3.1-23
3.1-24	3.1-24
3.1-25	3.1-25
3.1-26	3.1-26
3.3-65	3.3-65
3.3-70	3.3-70
3.6-44	3.6-44
3.6-45	3.6-45
3.6-47	3.6-47
3.6-49	3.6-49
3.6-51	3.6-51
3.6-52	3.6-52
3.6-53	3.6-53
3.7-9	3.7-9
3.7-10	3.7-10
3.7-11	3.7-11
- - -	3.9-22
- - -	3.9-23
- - -	B3.9-36
- - -	B3.9-37
- - -	B3.9-38

handling accident (FHA) analysis had been performed for all three BFN units. At that time, TVA stated that since the three units were essentially identical, comparable results would be expected for all three units. TVA stated that the existing License Condition 2.C.(4) provided a standing obligation for TVA to submit the remaining analyses for review prior to Unit 1 entering Mode 3 or above. This was affirmed by TVA in its letter of December 9, 2002. By letter dated May 17, 2004, TVA submitted descriptions of the Unit 1 analysis methods, assumptions, inputs, and results, thereby satisfying this obligation.

In a letter dated July 11, 2003, TVA requested an exemption from the requirements of 10 CFR 50, Appendix A, General Design Criterion (GDC) 41, Containment atmosphere cleanup. The NRC staff has determined that the exemption request is not required for the approval for full implementation of AST on the BFN units. However, the information provided in the exemption request was used by the NRC staff to support the technical evaluation discussed below.

The NRC staff reviewed all the supplements. The supplements augmented/withdrew portions of the submittal. The NRC staff determined that although the scope had been modified the originally published no significant hazards consideration determination (67 FR 63697) did not change. However a new *Federal Register* Notice (69 FR 22883) was issued to address the modifications to the submittal not originally noticed.

2.0 REGULATORY EVALUATION

In the past, power reactor licensees have typically used U.S. Atomic Energy Commission Technical Information Document (TID)-14844, Calculation of Distance Factors for Power and Test Reactor Sites, dated March 23, 1962, as the basis for DBA analysis source terms. The power reactor siting regulation, which contains offsite dose limits in terms of whole body and thyroid dose, 10 CFR 100.11, Determination of Exclusion Area, Low Population Zone, and Population Center Distance, makes reference to TID-14844.

In December 1999, the NRC issued a new regulation, 10 CFR 50.67, Accident Source Term, which provided a mechanism for licensed power reactors to replace the traditional accident source term used in their DBA analyses with an alternative source term. Section 50.67 of 10 CFR requires a licensee seeking to use an AST to apply for a license amendment and requires that the application contain an evaluation of the consequences of affected DBAs. Regulatory guidance for the implementation of these ASTs is provided in Regulatory Guide (RG) 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors. TVA 's application of July 2002, as supplemented, addresses these requirements in proposing to use the AST described in RG 1.183 as the DBA source term used to evaluate the radiological consequences of DBAs for BFN Units 1, 2 and 3. As part of the implementation of the AST, the total effective dose equivalent (TEDE) acceptance criterion of 10 CFR 50.67(b)(2) replaces the previous whole body and thyroid dose guidelines of 10 CFR 100.11 and 10 CFR Part 50, Appendix A, GDC 19, for the loss-of-coolant accident (LOCA), the main steam line break (MSLB) accident, the fuel handling accident (FHA), and the control rod drop accident (CRDA).

Part 50 of 10 CFR, Appendix A, GDC 26, requires that each reactor have two independent reactivity control systems of a different design, while GDC 29 requires that the reactivity control system be capable of accomplishing its safety function in the event of anticipated operational occurrences.

Section 50.49 of 10 CFR, Environmental Qualification of Equipment, requires that the safety-related electrical equipment which are relied upon to remain functional during and following the design basis events be qualified for accident (harsh) environment. This provides assurance that the equipment needed in the event of an accident will perform its intended function. Regulatory Position 1.3.5, 6, and Appendix I of RG 1.189 addresses the requirements for assessing the impact of the difference in source term characteristics on environmental qualification (EQ) doses. NUREG-1465, Accident Source Terms for Light-Water Nuclear Power Plants, provides estimates of AST that were more physically based and that could be applied to a BWR [boiling water reactor]. NUREG-0933 Issue 187, The Potential Impact of Postulated Cesium Concentration on Equipment Qualification1, indicated that for equipment exposed to the containment atmosphere, the TID-14844 source term and the gap and in-vessel releases in the AST produced similar integrated doses, and for equipment exposed to suppression pool water, the integrated doses calculated with the AST remain enveloped by those calculated with TID-14844 for the first 145 days post accident for a BWR, including the 30 percent vs. 1-percent release of cesium. It was concluded that there was no clear basis for back fitting the requirement to modify the design basis for equipment qualification to adopt the AST. There would be no discernible risk reduction associated with such a requirement.

NUREG-800, Standard Review Plan (SRP), Section 6.5.2, Containment Spray as a Fission Product Cleanup System, provides the acceptance criteria regarding the systems used to minimize iodine re-evolution as presented in the licensee's re-analysis of the radiological consequences for the LOCA. The BFN units were not licensed to many of the GDC contained in 10 CFR Part 50, Appendix A, but Section 1.5.1.6 of the Updated Final Safety Analysis Report (UFSAR) contains criteria that are essentially equivalent to GDC's. Maintaining compliance with the intent of these criteria was evaluated as part of the evaluation process.

On March 14, 2000, the NRC staff issued an amendment for Units 2 and 3 to increase the allowable main steam isolation valve (MSIV) leakage rate. This amendment permitted Units 2 and 3 to use the main steam drain lines to direct any MSIV leakage to the main condenser. This drain path takes advantage of the large volume of the main steam lines (MSLs) and condenser to provide holdup and plate-out of fission products that may leak through the closed MSIVs. The licensee performed evaluations and seismic verification walkdowns to demonstrate that the main steam system piping and components which comprise the alternate leakage treatment (ALT) system were seismically rugged and are able to perform the safety function of an MSIV leakage treatment system. By letter dated July 9, 2004, the licensee requested an amendment similar to that granted on Units 2 and 3. The licensee also submitted, in letters dated July 2, August 24, and September 17, 2004, the evaluations, seismic verification walkdowns, and seismic ruggedness evaluations to support the AST use of the ALT MSIV leak path for Unit 1. The seismic ruggedness evaluation was performed to demonstrate the seismic adequacy of the turbine building which houses the ALT system. The structural integrity of the turbine building is an important consideration to the adequacy of the alternate MSIV leakage path because a non-seismically designed turbine building should be capable of withstanding the earthquake without degrading the capability of the ALT system.

The licensee referenced the General Electric Company (GE) Report, NEDC-31858P-A, Boiling Water Reactor Owners Group (BWROG) Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems, Revision (Rev.) 2 (BWROG Report or NEDC-31858P), as a basis for the acceptability of its proposed license amendment. The BWROG report summarizes data on the seismic performance of main steam piping and

Table 1

BFN Accident Analysis Parameters

All Accidents	
Reactor power (3952 x 1.02), MWt	4031
SGTS Flow, cfm	
Stack, Elevated	24750
Damper bypass, ground level	10
SGTS drawdown time, min	<2
SGTS HEPA filter efficiency, particulate, %	90
SGTS Charcoal Filter Efficiency, %	Not credited
Dose conversion factors	FGR11/FGR12
Breathing rate, offsite, m^3/s	
0-8 hours	3.5E-4
8-24 hours	1.8E-4
>24 hours	2.3E-4
Breathing rate, control room,m^3/s	3.5E-4
Control room normal intake flow, cfm	6717
Control room unfiltered infiltration, cfm	3717
Control room filtered pressurization, cfm	3000
Control room volume, ft^3	210,000
Control room intake HEPA filter efficiency, particulate, %	90
Control room charcoal filter efficiency, %	Not credited
Control room occupancy factor	
0-24 hrs	1.0
1-4 days	0.6
4-30 days	0.4
Control Rod Drop Accident (RDA)	
Radial peaking factor	1.5
Fraction of core Inventory in gap	
Noble gases	0.1
Iodine	0.1
Br	0.05
Cs,Rb	0.12
Failed rods	850
Fraction of failed rods that reach melt	0.0077

Melted fuel release fraction to vessel	
Noble gases	1.0
Iodine	0.5
Br	0.3
Cs,Rb	0.25
Te	0.05
Ba, Sr	0.02
Noble metals	0.0025
Ce	0.0005
La	0.0002
Fraction of activity released to vessel that enters main condenser	
Noble gases	1.0
Iodine	0.1
others	0.01
Fraction of activity released from main condenser	
Noble gases	1.0
Iodine	0.1
others	0.01
Main condenser (plus LP turbine) free volume, ft^3	187,000
Release rate from main condenser, cfm	1850
Release duration, days	30

Loss of Coolant Accident

Containment Leakage Source

Onset of gap release phase, min	2.0

Core release fractions and timing–CNMT atmosphere

Duration, hrs	0.5000E+00	0.1500E+01
Noble Gases:	0.5000E-01	0.9500E+00
Iodine:	0.5000E-01	0.2500E+00
Cesium:	0.5000E-01	0.2000E+00
Tellurium:	0.0000E+00	0.5000E-01
Strontium:	0.0000E+00	0.2000E-01
Barium:	0.0000E+00	0.2000E-01
Noble Metals:	0.0000E+00	0.2500E-02
Cerium:	0.0000E+00	0.5000E-03
Lanthanum:	0.0000E+00	0.2000E-03

Iodine species distribution	
Elemental	0.0485
Organic	0.0015
Particulate	0.9500

Main condenser volume, ft^3	122,400
Primary CNMT volume, ft^3	
Drywell	159,000
Supression pool air space	119,400
CNMT leakrate, %/day	2.0
Unit 1 Secondary containment volume (50% of free volume)	1,311,209
Unit 2 and 3 Secondary containment volume (50% of free volume)	1,931,500
Hardened wet well vent release (elevated), 8 hours to 30 days, scfh	10
Release via SGTS (elevated) and base of stack (ground)	
SGTS ground level leakage (base of stack), cfm	10
Volume at base of stack (50% of free volume), ft^3	34,560
Drywell natural deposition	
Particulate	Powers 10%-percentile Model
Elemental	Same as particulate
Surface area for elemental iodine deposition in drywell, m^2	3409
Drywell maximum accident conditions	
Pressure, psig	48.3
Temperature, degF	294.9
Control room isolation delay, minutes	10
<u>CAD System Release</u>	
Activity same as CNMT leakage case	
Flow rate, cfm	139
CAD operation, days post accident	10, 20, 29
CAD operation duration, hours	24
No mixing in RB, release via elevated release point	
<u>MSIV Leakage</u>*	
Activity same as CNMT leakage case	
MSIV TS leak rate @25 psig, scfh	
One line	100
Total	150

Main steam line configuration for deposition analysis
- all four steam lines intact, in service at start of event
- One inboard MSIV fails to close
- In each of three isolated lines, a well-mixed control volume exists
- Only horizontal lines are credited
- 100 scfh is assumed to exist in faulted line
- One of remaining lines is assumed to leak at 50 scfh; other two are leaktight
- Pressure between closed MSIVs is assume to be equal to CNMT pressure
- Temperature is assumed to be normal steam line conditions
- Pressure downstream of outboard MSIVs (and inboard MSIV on faulted line) is assumed

to be atmospheric; normal operating temperature
MSIV leakage at test pressure is converted to volumetric flow based on post-LOCA drywell temperature and pressure
RADTRAD Bixler model used for elemental iodine
MSIV leakage from condenser is released without dilution or holdup in turbine building

MSIV Leakage that bypasses main condenser,% of total		0.5
Steam line deposition	Aerosol	Elemental
Steam line	99.87	99.01
MC bypass	89.33	16.37

ECCS Leakage

Iodine species fraction	Sump
Particulate/aerosol	0
Elemental	97
Organic	3
Suppression pool liquid volume, ft^3	121,500
Estimated leakage, gpm	5
Iodine Flash Fraction	0.1
SGTS charcoal filtration	Not credited

Release via SGTS (elevated) and base of stack (ground)

Fuel Handling Accident

Radial peaking factor	1.5
Fuel rods damaged *(conservatively based on 7 x 7 fuel)*	111
Decay period, hrs	24
Fraction of core in gap	
I-131	0.08
Kr-85	0.1
Other iodines	0.05
Other noble gases	0.05
Pool decontamination factor	200
Release period	Instantaneously
Hold-up and mitigation	No credit
Release via:	RB refueling zone vent

Main Steam Line Break

Reactor coolant activity, uCi/gm dose equivalent I-131	
Normal	3.2
Spike	32
Mass release, lbm	
Steam	11,975
Liquid (saturated at 898 psia)	42,215
Release flash fraction (pressure=1020 psia)	0.38
Release duration, sec	5.5
Iodine species	100% Elemental