Tennessee Valley Authority, 1101 Market Street, Chattanooga, Tennessee 37402

CNL-16-045

March 9, 2016

10 CFR 50.90

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Browns Ferry Nuclear Plant, Units 1, 2, and 3
Renewed Facility Operating License Nos. DPR-33, DPR-52, and DPR-68
NRC Docket Nos. 50-259, 50-260, and 50-296

Subject: **Proposed Technical Specifications (TS) Change TS-505 - Request for License Amendments - Extended Power Uprate (EPU) - Supplement 6, Response to Request for Additional Information**

References:

1. Letter from TVA to NRC, CNL-15-169, "Proposed Technical Specifications (TS) Change TS-505 - Request for License Amendments - Extended Power Uprate (EPU)," dated September 21, 2015 (ML15282A152)

2. Letter from NRC to TVA, "Browns Ferry Nuclear Plant, Units 1, 2, and 3 - Request for Additional Information Related to License Amendment Request Regarding Extended Power Uprate (CAC Nos. MF6741, MF6742, and MF6743)," dated February 26, 2016 (ML16040A232)

3. Letter from NRC to TVA, "Browns Ferry Nuclear Plant, Units 1, 2, and 3 - Request for Additional Information Related to License Amendment Request Regarding Extended Power Uprate (CAC Nos. MF6741, MF6742, and MF6743)," dated February 29, 2016 (ML16049A463)

By the Reference 1 letter, Tennessee Valley Authority (TVA) submitted a license amendment request (LAR) for the Extended Power Uprate (EPU) of Browns Ferry Nuclear Plant (BFN) Units 1, 2 and 3. The proposed LAR modifies the renewed operating licenses to increase the maximum authorized core thermal power level from the current licensed thermal power of 3458 megawatts to 3952 megawatts. During the technical review of the LAR, the NRC identified the need for additional information. The Reference 2 and 3 letters provided NRC Requests for Additional Information (RAIs) related to the dose analyses and radiation protection, respectively. The due date for the responses to the NRC RAIs provided by the Reference 2 letter is March 18, 2016. The due date for the responses to the NRC RAIs provided by the Reference 3 letter is April 1, 2016. Enclosure 1 to this letter provides the responses to the dose analyses RAIs included in the Reference 2 letter. Enclosure 2 to this letter provides the responses to the radiation protection RAIs included in the Reference 3 letter.

TVA has reviewed the information supporting a finding of no significant hazards consideration and the environmental consideration provided to the NRC in the Reference 1 letter. The supplemental information provided in this submittal does not affect the bases for concluding that the proposed license amendment does not involve a significant hazards consideration. In addition, the supplemental information in this submittal does not affect the bases for concluding that neither an environmental impact statement nor an environmental assessment needs to be prepared in connection with the proposed license amendment. Additionally, in accordance with 10 CFR 50.91(b)(1), TVA is sending a copy of this letter to the Alabama State Department of Public Health.

There are no new regulatory commitments associated with this submittal. If there are any questions or if additional information is needed, please contact Mr. Edward D. Schrull at (423) 751-3850.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 9th day of March 2016.

Respectfully,

J. W. Shea

Digitally signed by J. W. Shea
DN: cn=J. W. Shea, o=Tennessee Valley Authority, ou=Nuclear Licensing, email=jwshea@tva.gov, c=US
Date: 2016.03.09 20:22:14 -05'00'

J. W. Shea
Vice President, Nuclear Licensing

Enclosures:

1. Response to NRC Requests for Additional Information ARCB-DA-RAI 1, ARCB-DA-RAI 2, ARCB-DA-RAI 3, and ARCB-DA-RAI 4

2. Response to NRC Request for Additional Information ARCB-RP-RAI 1, ARCB-RP-RAI 2, ARCB-RP-RAI 3, and ARCB-RP-RAI 4

cc:

NRC Regional Administrator - Region II
NRC Senior Resident Inspector - Browns Ferry Nuclear Plant
State Health Officer, Alabama Department of Public Health

ENCLOSURE 1

Response to NRC Requests for Additional Information
ARCB-DA-RAI 1, ARCB-DA-RAI 2, ARCB-DA-RAI 3, and ARCB-DA-RAI 4

ENCLOSURE 1

ARCB-DA-RAI 1

In an effort to ensure a complete and accurate review of the dose consequence analyses, please provide additional information (preferably in tabular form) describing, for each DBA affected by the proposed extended power uprate (EPU), all the basic parameters and assumptions used in the dose consequence analyses (See Issue 1 of NRC Regulatory Issue Summary (RIS) 2006-04, "Experience with Implementation of Alternative Source Terms" (ADAMS Accession No. ML053460347). Please provide the current licensing basis (CLB) and the revised EPU input values, assumptions, and methods, as well as a justification for any changes to the CLB. Please identify which of these parameters were not previously reviewed and approved by the NRC and provide a justification for the change from the previously reviewed values to the CLB.

The NRC staff notes that some of the requested information has been provided in textual form in Section 2.9.2, "Radiation Sources in Reactor Coolant," of NEDC-33860P, Revision 0, "Safety Analysis Report for Browns Ferry Nuclear Power Plant, Units 1, 2, and 3, Extended Power Uprate," dated September 21, 2015, provided in Attachment 6 of the LAR (hereafter "Section 2.9.2"). The NRC staff requests that the information in Section 2.9.2 be expanded to include all of the basic parameters, whether or not the individual parameter is being changed for the EPU amendment. The staff also finds it helpful if the information is presented in separate tables for each affected accident (loss-of-coolant accident (LOCA), control rod drop accident (CRDA), main steamline break accident (MSLBA), and fuel handling accident (FHA)). Please state if each accident's methods, inputs, assumptions, and results have not changed from those reviewed and approved by the NRC in "Reference 72" ["Browns Ferry Nuclear Plant, Units 1, 2, and 3 - Issuance of Amendments Regarding Full-Scope Implementation of Alternative Source Term (TAC Nos. MB5733, MB5734, MB5735, MC0156, MC0157 and MC0158) (TS-405)," September 27, 2004 (ADAMS Accession No. ML042730028)].

TVA Response:

As requested, the input parameters, assumptions, and methods used in the dose consequence analysis for each Design Basis Accident (DBA) have been tabulated in Tables 1 through 5. These tables include a comparison to parameters/assumptions previously reviewed and approved in “Reference 72” [“Browns Ferry Nuclear Plant, Units 1, 2, and 3 - Issuance of Amendments Regarding Full-Scope Implementation of Alternative Source Term (TAC Nos.MB5733, MB5734, MB5735, MC0156, MC0157 and MC0158) (TS-405),” September 27, 2004 (Reference 1 of this RAI response)] and TS-474 (Reference 2) to the EPU LAR submittal. The Loss of Coolant Accident (LOCA), Control Rod Drop Accident (CRDA), Fuel Handling Accident (FHA), and Main Steam Line Break Accident (MSLBA) analyses were performed in accordance with the Regulatory Guide 1.183 (Reference 3) guidance detailed previously in Reference 72 (Reference 1 of this RAI response). The LOCA analysis was subsequently revised and approved in the license amendments issued for Technical Specification (TS) Change Request TS-474 (Reference 2) as discussed below.

On August 27, 2010, Tennessee Valley Authority (TVA) submitted a Technical Specifications (TS) change, TS-474 (Reference 4), to modify TS 3.7.3 to permit one or more Control Room Envelope Ventilation System (CREVS) subsystems to be inoperable for 90 days. To support this license amendment request (LAR), the LOCA (the only accident to credit CREVS) was evaluated to determine the combined impact of no credit for CREVS and the increased Engineering Safeguard Feature (ESF) leakage on the radiological dose to operators in the BFN control rooms. This analysis resulted in an increase in the CLB control room TEDE dose from 1.62 rem to 1.94 rem (See Table 1 in Reference 5 for specific details used in the new analysis).

The EPU LOCA dose analysis results were originally submitted August 27, 2010 (Reference 4), and supplemented by References 5 and 6. TS-474 was approved on July 30, 2012 (Reference 2). Thus, the EPU-presented LOCA analysis has been reviewed and approved by the NRC.

Table 6 summarizes the differences in the results of the DBA that was approved in Reference 72 (Reference 1 of this RAI response) and the EPU LAR. Note that the differences in the LOCA analysis listed below were reviewed for the TS-474 change (Reference 4) and approved on July 30, 2012 (Reference 2). The changes in the dose due the changes to the CRDA inputs constitute a less than minimal change in the doses at the offsite receptor and at the control room.

<u>LOCA (See Tables 1 and 5 for inputs and Table 6 for dose results)</u>

- Base of stack leakage increased from 10 cfm to 20 cfm revised to provide additional margin for damper testing.
- Credit for the Control Room Envelope Ventilation System (CREVS) was removed for conservatism in support of Technical Specification Change Request TS-474.
  - Control Room Makeup Filtered Flow changed from 3000 cfm to 0 cfm.
  - Control Room Makeup Filtered Flow filtration credit changed to no filtration credit.
  - Control Room Unfiltered Inleakage increased from 3717 cfm to 6717 cfm.
  - CREVS HEPA filtration credit changed to no HEPA filtration credit.
- Emergency Core Cooling System Leakage into the Reactor Building changed from 10 gpm to 20 gpm for conservatism.
- Unit 2 and 3 Reactor Building volume changed to Unit 1 Reactor Building volume because the small volume was limiting (i.e., less holdup) and is applicable to all units.
- The Unit 1 Turbine Building Exhaust Release atmospheric dispersion factors (X/Qs) to the control room intakes were used for conservatism. These X/Qs bound the X/Qs for all Turbine Building release points for all units to the control room intakes for the Main Steam Isolation Valve (MSIV) leakage pathway.

  There are two release points from the Turbine Building. One is for the "Turbine Building Roof Ventilators" (applicable to all units); the second is for the "Turbine Building Exhaust Release" which is uniquely calculated for each unit. For Units 2 and 3, the roof ventilator set bounds the exhaust release set; the same is not true for Unit 1. Therefore, the LOCA dose analysis uses the bounding set of Turbine Building X/Qs (i.e., Unit 1 Turbine Building Exhaust Release X/Qs to the control room intakes) for the MSIV leakage pathway.

(For details see Response to ARCB-DA-RAI 3)

<u>CRDA (See Tables 2 and 5 for inputs and Table 6 for results)</u>

- Base of stack leakage increased from 10 cfm (AST) to 20 cfm (EPU) revised to provide additional margin for damper testing.

<u>FHA (See Tables 3 and 5 for inputs and Table 6 for results)</u>

There were no differences.

<u>MSLBA (See Table 4 for inputs and Table 6 for results)</u>

There were no differences.

Table 6 summarizes the differences in the results of the DBA radiological consequences listed in the AST submittal (Reference 7) and approved in Reference 1 compared to the TS-474 submittal (Reference 4) approved in Reference 2 for the LOCA and any intermediate analyses performed under the auspices of 10 CFR 50.59. The DBA radiological consequences described in NEDC-33860, Section 2.9.2, of the EPU LAR used the same methodology contained in Regulatory Guide 1.183 (Reference 3) and approved as part of the AST amendment (Reference 1).

References

1. Letter from NRC to TVA, “Browns Ferry Nuclear Plant, Units 1, 2, and 3 - Issuance of Amendments Regarding Full-Scope Implementation of Alternative Source Term (TAC Nos. MB5733, MB5734, MB5735, MC0156, MC0157 and MC0158) (TS-405),” dated September 27, 2004 (ML042730028).

2. Letter from NRC to TVA, “Browns Ferry Nuclear Plant, Units 1, 2, and 3 - Issuance of Amendments Regarding Request to Add Technical Specification 3.7.3, ‘Control Room Emergency Ventilation (CREV) System’ (TAC Nos. ME4668, ME4669, and ME4670) (TS-474),” dated July 30, 2012 (ML11189A217).

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

4. Letter from TVA to NRC, “Technical Specifications Change TS-474 - Request to Add a TS 3.7.3, ‘Control Room Emergency Ventilation (CREV) System,’ Action to Address Two CREV Subsystems Inoperable Due to Inoperable CREV System High Efficiency Particulate Air (HEPA) Filter and/or Charcoal Adsorbers,” dated August 27, 2010 (ML102430528).

5. Letter from TVA to NRC, “Response to Request for Additional Information Regarding Technical Specifications Change TS-474 - TS 3.7.3, ‘Control Room Emergency Ventilation System,’ dated March 10, 2011,” dated April 11, 2011 (ML11105A151).

6. Letter from TVA to NRC, “Supplement to Technical Specifications Change TS-474 - Request to Add a TS 3.7.3, "Control Room Emergency Ventilation (CREV) System," Action to Address Two CREV Subsystems Inoperable Due to Inoperable CREV System High Efficiency Particulate Air (HEPA) Filter and/or Charcoal Adsorbers,” dated January 13, 2012 (ML12017A161).

7. Letter from TVA to NRC, “Browns Ferry Nuclear Plant (BFN) – Units 1, 2, and 3 – License Amendment – Alternative Source Term,” dated July 31, 2002 (ML022200382).

List of Acronyms for Tables

AST	Alternative Source Term
BFN	Browns Ferry Nuclear Plant
CAD	Containment Atmospheric Dilution
CREVS	Control Room Emergency Ventilation System
cfm	cubic feet per min
ECCS	Emergency Core Cooling System
EPU	Extended Power Uprate
ft^3	cubic feet
gpm	gallons per minute
hrs	hours
HEPA	High Efficiency Particulate Air
LAR	License Amendment Request
min	minute
m^3	cubic meter
MWt	Mega watt thermal
MSIV	Main Steam Isolation Valve
MVP	Mechanical Vacuum Pump
RB	Reactor Building
scfh	Standard cubic foot per hour
SGTS	Standby Gas Treatment System
TEDE	Total Effective Dose Equivalent
μCi	micro curie

Table 1 BFN-1, 2, and 3 Accident Analysis Parameters - LOCA Inputs

<table>
<tr><th>Parameter</th><th colspan="3">Values [(a)]</th></tr>
<tr><td>Reactor Power (3952 x 1.02) MWt</td><td colspan="3">4031</td></tr>
<tr><td colspan="4">Containment Leakage Source</td></tr>
<tr><td>Fission Product Release Timing (min)
Gap Release
Early In-Vessel</td><td colspan="3">
2 to 32
32 to 122</td></tr>
<tr><td rowspan="11">Core release fractions and timing–Containment atmosphere</td><td>Duration, hrs</td><td>5.00E-01</td><td>1.50E+00</td></tr>
<tr><td>Radionuclide Group</td><td>Gap</td><td>Early In-Vessel</td></tr>
<tr><td>Noble Gases</td><td>5.00E-02</td><td>9.50E-01</td></tr>
<tr><td>Iodine</td><td>5.00E-02</td><td>2.50E-01</td></tr>
<tr><td>Alkali Metals</td><td>5.00E-02</td><td>2.00E-01</td></tr>
<tr><td>Tellurium Group</td><td>0.00E+00</td><td>5.00E-02</td></tr>
<tr><td>Strontium</td><td>0.00E+00</td><td>2.00E-02</td></tr>
<tr><td>Barium</td><td>0.00E+00</td><td>2.00E-02</td></tr>
<tr><td>Noble Metals</td><td>0.00E+00</td><td>2.50E-03</td></tr>
<tr><td>Cerium Group</td><td>0.00E+00</td><td>5.00E-04</td></tr>
<tr><td>Lanthanum Group</td><td>0.00E+00</td><td>2.00E-04</td></tr>
<tr><td>Iodine species distribution
Elemental
Organic
Particulate</td><td colspan="3">
4.85%
0.15%
95%</td></tr>
<tr><td>Main condenser volume, ft^3</td><td colspan="3">122,400</td></tr>
<tr><td>Primary Containment volume, ft^3
Drywell
Suppression pool air space</td><td colspan="3">
159,000
119,400</td></tr>
<tr><td>Containment leak rate, %/day</td><td colspan="3">2.0</td></tr>
<tr><td>Secondary containment volume (50% of free volume)[(b)]</td><td colspan="3">1,311,209</td></tr>
<tr><td>Hardened wet well vent release (elevated), 8 hours to 30 days, scfh</td><td colspan="3">10</td></tr>
<tr><td>SGTS Flow, cfm
Stack, Elevated
Damper bypass, ground level [(b)]</td><td colspan="3">
24750
20</td></tr>
<tr><td>Volume at base of stack (50% of free volume), ft^3</td><td colspan="3">34,560</td></tr>
<tr><td>Drywell natural deposition
Particulate
Elemental</td><td colspan="3">
Powers 10%-percentile Model
Same as particulate</td></tr>
<tr><td>Surface area for elemental iodine deposition in drywell, m^2 (ft^2)</td><td colspan="3">316.7 (3409)</td></tr>
<tr><td colspan="4">CAD System Release</td></tr>
<tr><td>Flow rate, cfm (from torus to stack room)</td><td colspan="3">139</td></tr>
<tr><td>CAD operation, days post accident</td><td colspan="3">10, 20, 29</td></tr>
<tr><td>CAD operation duration, hours</td><td colspan="3">24</td></tr>
<tr><td colspan="4">No mixing in the RB, release via elevated release point.</td></tr>
</table>

Parameter	**Values** [(a)]
MSIV Leakage	
Activity same as Containment leakage case	
MSIV TS leak rate @25 psig, scfh One line Total	 100 150
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 leak tight • 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 Steam line Main condenser bypass	Aerosol / Elemental Iodine 99.87 / 99.01 89.33 / 16.37
ECCS leakage	
Iodine species fraction Particulate/aerosol Elemental Organic	Sump 0 97 3
Suppression pool liquid volume, ft^3	121,500
Estimated leakage, gpm [(b)]	20
Iodine Flash Fraction	0.1
SGTS HEPA filtration efficiency, %	90
SGTS charcoal filtration	Not credited
SGTS Flow, cfm Stack, Elevated Damper bypass, ground level	 24750 20
Control Room	
Control room normal intake flow, cfm	6717[(b)]
Control room unfiltered infiltration, cfm	6717[(b)]
Control room filtered pressurization, cfm	0
Control room volume, ft^3	210,000
Control room isolation delay, minutes	10
Control Room Occupancy Factors	0-1 day: 1.0 1-4 days: 0.6 4-30 days: 0.4
Breathing rate, control room,m^3/s	3.5E-4
CREVS charcoal filter efficiency, %	Not credited
CREVS HEPA filter particulate efficiency, %	Not credited [(b)]
General Inputs	
Dose conversion factors	FGR11/FGR12
Breathing rate, offsite, m^3/s 0-8 hours 8-24 hours	 3.5E-4 1.8E-4

Parameter	**Values** [a]			
> 24 hours	2.3E-4			
Atmospheric Dispersion Factors (sec/m^3)				
Receptors	**Control Room** [c]		**Site Boundary**	
Release via Top of Stack				
Time Period (hrs)	**Unit 1**	**Unit 3**	**EAB**	**LPZ**
Fumigation Interval	3.40E-05	*	2.35E-05	1.26E-05
0-2	**	1.41E-07	1.19E-06	1.13E-06
2-8	**	4.50E-08		5.75E-07
8-24	**	2.54E-08		4.10E-07
24-96	**	7.36E-09		1.97E-07
96-720	**	1.24E-09		6.88E-08
Release via Base of Stack				
0-2	2.00E-04	*	2.62E-04	1.31E-04
2-8	1.28E-04	*		6.61E-05
8-24	5.72E-05	*		4.69E-05
24-96	4.05E-05	*		2.23E-05
96-720	3.09E-05	*		7.96E-06
Release via Turbine Building Exhaust (i.e., MSIV Leakage) [b]				
0-2	3.22E-04	*	2.62E-04	1.31E-04
2-8	2.77E-04	*		6.61E-05
8-24	1.31E-04	*		4.69E-05
24-96	7.91E-05	*		2.23E-05
96-720	6.10E-05	*		7.96E-06

(a) All values used in this accident analysis are the same for AST SER and EPU conditions unless otherwise noted.
(b) This was approved in the TS-474 amendment.
(c) Due to the dual intake configuration for the control room, limiting X/Qs need to be divided by two.

*Unit 1 intake limiting.
**Unit 3 intake limiting.

Table 2 BFN-1, 2, and 3 Accident Analysis Parameters - CRDA Inputs

Parameter	Values [(a)]
Reactor Power (3952 x 1.02) MWt	4031
Radial peaking factor	1.5
Non-melted Fuel Release Fractions of core inventory in gap for radionuclide groups: Noble gases Iodine Other Halogens Cs, Rb	 0.1 0.1 0.05 0.12
Failed Fuel Rods	850
Fraction of failed fuel rods that reach melt	0.0077 of 850 failed fuel rods
Melted fuel release fractions to vessel for radionuclide groups: Noble gases Iodine Other Halogens Alkalis Metals Tellerium Group Barium, Strontium Noble metals Cerium group Lanthanum group	 1.0 0.5 0.3 0.25 0.05 0.02 0.0025 0.0005 0.0002
Fraction of activity released to vessel that enters main condenser Noble gases Iodine Others	 1.0 0.1 0.01
Fraction of activity released from main condenser Noble gases Iodine Others	 1.0 0.1 0.01
Iodine species fraction Particulate/aerosol Elemental Organic	 0 97 3
Main condenser (plus LP turbine) free volume, ft^3	187,000 (136,000 + 51,000)
MVP flow rate	1850 at 7" Hg
Condenser to Stack Room Leakage, cfm	20
Stack Room Free Volume, ft^3	34,560
Stack Room to Environment Leakage, cfm	20 [(b)]
SGTS stack filtration	Not credited
SGTS Flow, cfm Stack, Elevated Damper bypass, ground level	 24750 20
Control room normal intake flow, cfm	6717
Control room unfiltered infiltration, cfm	6717
Control room filtered pressurization, cfm	0
Control room volume, ft^3	210,000
Control Room Occupancy Factors	0-1 day: 1.0 1-4 days: 0.6 4-30 days: 0.4
Breathing rate, control room, m^3/s	3.5E-4
CREVS charcoal filter efficiency, %	Not credited
CREVS HEPA filter particulate efficiency, %	Not credited

ENCLOSURE 1

Parameter	Values [a]			
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			
Release duration, days	30 days			
Atmospheric Dispersion Factors (sec/m^3)				
Receptors	**Control Room [c]**		**Site Boundary**	
Release via Top of Stack				
Time Period (hrs)	**Unit 1**	**Unit 3**	**EAB**	**LPZ**
Fumigation Interval	3.40E-05	*	2.35E-05	1.26E-05
0-2	**	1.41E-07	1.19E-06	1.13E-06
2-8	**	4.50E-08		5.75E-07
8-24	**	2.54E-08		4.10E-07
24-96	**	7.36E-09		1.97E-07
96-720	**	1.24E-09		6.88E-08
Release via Base of Stack				
0-2	2.00E-04	*	2.62E-04	1.31E-04
2-8	1.28E-04	*		6.61E-05
8-24	5.72E-05	*		4.69E-05
24-96	4.05E-05	*		2.23E-05
96-720	3.09E-05	*		7.96E-06

(a) All values used in this accident analysis are the same for AST SER and EPU conditions unless otherwise noted.
(b) Base of stack leakage increased from 10 cfm to 20 cfm revised to provide additional margin for damper testing.
(c) Due to the dual intake configuration for the control room, limiting X/Qs need to be divided by two.

*Unit 1 intake limiting.

**Unit 3 intake limiting.

Table 3 BFN-1, 2, and 3 Accident Analysis Parameters - FHA Inputs

Parameters	**Values [(a)]**			
Reactor Power (3952 x 1.02) MWt	4031			
Radial peaking factor	1.5			
Fuel rods damaged (conservatively based on 7x7 fuel)	111			
Decay period, hours	24			
Fraction of core in gap I-131 Kr-85 Other iodines Other noble gases	 0.08 0.10 0.05 0.05			
Overall effective pool decontamination factor	200			
Release via	RB refueling zone vent			
Control room normal intake flow, cfm	6717			
Control room unfiltered infiltration, cfm	6717			
Control room filtered pressurization, cfm	0			
Control room volume, ft^3	210,000			
Control Room Occupancy Factors	0-1 day: 1.0 1-4 days: 0.6 4-30 days: 0.4			
Breathing rate, control room, m^3/s	3.5E-4			
CREVS charcoal filter efficiency, %	Not credited			
CREVS HEPA filter particulate efficiency, %	Not credited			
Dose conversion factors	FGR11/FGR12			
Breathing rate, offsite, m^3/s 0-8 hours 8-24 hours > 24 hours	 3.5E-4 1.8E-4 2.3E-4			
Release Period	Instantaneous			
Holdup and migration credited	No credited			
Atmospheric Dispersion Factors (sec/m^3)				
Receptors	**Control Room [(b)]**		**Site Boundary**	
Release via Reactor Building refueling zone vent				
Time Period (hrs)	**Unit1**	**Unit 3**	**EAB**	**LPZ**
0-2	4.60E-04	*	2.62E-04	1.31E-04

(a) All values used in this accident analysis are the same for AST SER and EPU conditions unless otherwise noted.
(b) Due to the dual intake configuration for the control room, limiting X/Qs need to be divided by two.

*Unit 1 intake limiting.

Table 4 BFN-1, 2, and 3 Accident Analysis Parameters - MSLBA Inputs

Parameters	**Values** [(a)]			
Reactor coolant activity[(b)], μCi/gm dose equivalent I-131 Normal Pre-Accident Spike	 3.2 32			
Mass release, lbm Steam Liquid (saturated at 898 psia)	 11,975 42,215			
Flashing fraction	0.3576 (pressure = 898 psia)			
Steam release duration, seconds	5.5			
CREVS Charcoal and HEPA filtration	Not credited			
Holdup, dilution, condensation, plateout, and sedimentation in Turbine Building	Not credited			
Iodine Species	100% elemental			
Atmospheric Dispersion Factors (sec/m^3)				
Receptors	**Control Room** [(c)]		**Site Boundary**	
Release via Turbine Building Exhaust				
Time Period (hrs)	**Unit 1**	**Unit 3**	**EAB**	**LPZ**
0-2	3.22E-04	*	2.62E-04	1.31E-04
2-8	2.77E-04	*		6.61E-05
8-24	1.31E-04	*		4.69E-05
24-96	7.91E-05	*		2.23E-05
96-720	6.10E-05	*		7.96E-06

(a) All values used in this accident analysis are the same for AST SER and EPU conditions unless otherwise noted.
(b) Reactor coolant activities are based on ANSI/ANS-18.1-1984 modified to reflect the dose equivalent I-131 of 3.2 μCi/gm at normal operating conditions (based on BFN Technical Specifications Section 3.4.6) and the 32 μCi/gm for the iodine spike.
(c) Due to the dual intake configuration for the control room, limiting X/Qs need to be divided by two.

*Unit 1 intake limiting.

Table 5 BFN-1, 2, and 3 Core Inventory

Activity (Ci/MWt)						
Nuclide	**t=0 hr**	**t=24 hr**		**Nuclide**	**t=0 hr**	**t=24 hr**
Co58	1.430E+02	1.416E+02		Xe131M	3.544E+02	3.487E+02
Co60	1.425E+02	1.424E+02		Te132	3.829E+04	3.089E+04
Kr83M	3.432E+03	1.387E+01		I132	3.885E+04	3.184E+04
Kr85	3.601E+02	3.601E+02		I133	5.534E+04	2.559E+04
Kr85M	7.329E+03	1.811E+02		Xe133	5.504E+04	5.303E+04
Rb86	6.372E+01	6.141E+01		Xe133M	1.734E+03	1.562E+03
Kr87	1.446E+04	3.051E-02		I134	6.141E+04	1.450E-03
Kr88	2.009E+04	5.743E+01		Cs134	5.703E+03	5.697E+03
Kr89	2.521E+04	0.000E+00		I135	5.250E+04	4.189E+03
Sr89	2.786E+04	2.748E+04		Xe135	1.971E+04	1.429E+04
Sr90	3.165E+03	3.165E+03		Xe135M	1.135E+04	6.823E+02
Y90	3.283E+03	3.273E+03		Cs136	1.941E+03	1.841E+03
Sr91	3.487E+04	6.103E+03		Xe137	5.023E+04	0.000E+00
Y91	3.583E+04	3.564E+04		Cs137	4.037E+03	4.037E+03
Sr92	3.677E+04	7.922E+01		Ba137M	3.829E+03	3.810E+03
Y92	3.696E+04	1.168E+03		Xe138	4.757E+04	1.172E-26
Y93	4.147E+04	8.084E+03		Ba139	4.930E+04	4.170E-01
Zr95	4.880E+04	4.822E+04		Ba140	4.909E+04	4.644E+04
Nb95	4.897E+04	4.897E+04		La140	5.231E+04	5.079E+04
Zr97	4.953E+04	1.851E+04		La141	4.498E+04	7.085E+02
Mo99	5.088E+04	3.956E+04		Ce141	4.535E+04	4.463E+04
Tc99M	4.454E+04	3.772E+04		La142	4.397E+04	1.035E+00
Ru103	4.094E+04	4.018E+04		Ce143	4.245E+04	2.597E+04
Ru105	2.710E+04	6.615E+02		Pr143	4.113E+04	4.075E+04
Rh105	2.559E+04	1.840E+04		Ce144	3.810E+04	3.810E+04
Ru106	1.488E+04	1.486E+04		Nd147	1.806E+04	1.698E+04
Sb127	2.796E+03	2.369E+03		Np239	5.201E+05	3.902E+05
Te127	2.773E+03	2.580E+03		Pu238	2.805E+02	2.805E+02
Te127M	3.721E+02	3.719E+02		Pu239	1.234E+01	1.238E+01
Sb129	8.457E+03	1.952E+02		Pu240	1.730E+01	1.730E+01
Te129	8.326E+03	1.236E+03		Pu241	4.450E+03	4.448E+03
Te129M	1.615E+03	1.590E+03		Am241	5.449E+00	5.470E+00
Te131M	5.155E+03	2.976E+03		Cm242	1.234E+03	1.234E+03
I131	2.669E+04	2.481E+04		Cm244	5.697E+01	5.697E+01

Table 6 Summary of BFN-1, 2, and 3 DBA Dose Consequences

	TEDE Dose (rem)		
	EAB	LPZ	Control Room
LOCA (AST LAR)	1.02	1.25	1.25
LOCA (10 CFR 50.59)	1.57	2.34	1.62
LOCA (EPU & TS-474)	1.71	2.38	1.94
Dose Limit	25	25	5
CRDA (AST LAR)	1.19	0.682	0.248
CRDA (50.59)	1.168	0.6958	0.2584+
Dose Limit	6.3	6.3	5
FHA	0.855	0.427	0.543
Dose Limit	6.3	6.3	5
MSLBA			
3.2 µCi/gm DEI-131	0.130	0.0652	0.0409
Dose Limit (Equilibrium Case)	2.5	2.5	5
32 µCi/gm DEI-131	1.30	0.652	0.409
Dose Limit (Iodine Spike)	25	25	5

+ Base of stack leakage increased from 10 cfm to 20 cfm. This value was revised to provide additional margin for damper testing.

ARCB-DA-RAI 2

The LAR stated that analysis methods were not changed from those used in "Reference 72" for the CRDA. The LAR also stated that the analysis was performed based on plant operation at the EPU power level of 3,952 MWt, and the updated design inputs were confirmed to remain applicable or bounding for the EPU conditions. Confirm that the revised analyses are performed at 102 percent of the proposed EPU power level. If the revised analyses are performed at the EPU power level (3,952 MWt), explain how Regulatory Position 3.1 (footnote 8) of RG 1.183 is met, or justify why an uncertainty factor is not used in the CRDA analysis.

TVA Response:

The Control Rod Drop Accident (CRDA) was performed at 102% of 3952 MWt (1.02 x 3952 = 4031 MWt) in accordance with RG 1.183, Regulatory Position 3.1 (footnote 8) (Reference 1).

Reference

1. NRC Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," dated July 2000.

ARCB-DA-RAI 3

Section 2.9.2 states that the LOCA analysis methods were not changed from those used in "Reference 72" and that all significant design-basis inputs and assumptions are the same as those in "Reference 72." The LOCA dose results provided by TVA in the application, and the supplements reviewed by the staff for "Reference 72," do not appear to match the LOCA consequences provided for the LOCA in the EPU LAR, Table 2.9-6, "LOCA Radiological Consequences." Therefore, it appears that the LOCA consequences have been updated since the NRC approval in "Reference 72." If the analysis methods, and all significant design-basis inputs and assumptions in the EPU LOCA analysis are the same as those in "Reference 72," explain why the EPU LOCA analysis consequences do not match the "Reference 72" LOCA analysis consequences.

TVA Response:

The Loss of Coolant Accident (LOCA) dose consequences were evaluated in accordance with Regulatory Guide 1.183, Appendix A (Reference 1). Some changes were made to the LOCA dose analysis since the Alternative Source Term (AST) submittal and subsequent approval by NRC. These changes were implemented to increase conservatism in selected input parameters and to support a subsequent License Amendment to the AST submittal.

As discussed in the response to ARCB-DA-RAI 1, Section 2.9.2 of NEDC-33860 should have included a reference to the Technical Specification change, TS-474, which was approved on July 30, 2012 (Reference 2 of this RAI response) for the LOCA analysis.

Thus, the Current Licensing Basis (CLB) LOCA analysis has been reviewed and approved by the NRC.

The design input and methodology used in the AST submittal are identical with the following exceptions that were reviewed and approved for TS-474:

(1) Credit for the control room envelope ventilation system (CREVS) was removed, i.e., the entire 6717 cfm make up flow into the control room from the environment is considered unfiltered.

    Previously, it was assumed that CREVS intake was 3000 cfm plus 3717 cfm of unfiltered inleakage. Then, once isolation occurred, the 3000 cfm intake was filtered and the 3717 cfm of unfiltered inleakage continued. For this submittal, the entire 6717 cfm is unfiltered.

(2) The allowable Standby Gas Treatment System flow bypassing the stack was increased from 10 cfm to 20 cfm.

(3) ESF leakage was increased from 2.5 gpm (conservatively analyzed as 5 gpm) to 10 gpm (conservatively analyzed as 20 gpm).

(4) The Reactor Building volume for Unit 1 was used rather than the Unit 2 and 3 Reactor Building volumes (1.311E+06 ft^3 vs 1.932E+06 ft^3, respectively) because the smaller reactor volume was found to be more limiting, i.e. a smaller volume would yield less holdup.

(5) The Unit 1 Turbine Building Exhaust Release atmospheric dispersion factors (X/Qs) to the control room intakes were used for conservatism. These X/Qs bound the X/Qs for all Turbine Building release points for all units for the Main Steam Isolation Valve (MSIV) leakage pathway.

There are two release points from the Turbine Building. One is for the "Turbine Building Roof Ventilators" (applicable to all units); the second is for the "Turbine Building Exhaust Release" which is uniquely calculated for each unit. For Units 2 and 3, the roof ventilator set bounds the exhaust release set; the same is not true for Unit 1. Therefore, the LOCA dose analysis uses the bounding set of Turbine Building X/Qs (i.e., Unit 1 Turbine Building Exhaust Release X/Qs to the control room intakes) for the Main Steam Isolation Valve (MSIV) leakage pathway.

References

1. NRC Regulatory Guide 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” dated July 2000.

2. Letter from NRC to TVA, “Browns Ferry Nuclear Plant, Units 1, 2, and 3 - Issuance of Amendments Regarding Request to Add Technical Specification 3.7.3, ‘Control Room Emergency Ventilation (CREV) System’ (TAC Nos. ME4668, ME4669, and ME4670) (TS-474),” dated July 30, 2012 (ML11189A217).

ARCB-DA-RAI 4

Section 2.9.2 states that the post-LOCA doses for the technical support center (TSC) were analyzed for the EPU, and the analyses methods were not changed from "Reference 72." Section 2.9.2 also states that the TSC is at the same location as the control room within the control room habitability zone. Thus, the same atmospheric dispersion factors were used to calculate the dose at the TSC receptor, but no other details on how the TSC doses were calculated are provided.

Please provide sufficient information regarding the methodology, inputs, and assumptions used to calculate the TSC doses so that the NRC staff can independently calculate the TSC doses for the CRDA, MSLBA, and FHA. Also, please provide a simplified diagram of the TSC ventilation system and explain in further detail the operation of the TSC, including the specific flow rates through the components during normal and accident conditions.

TVA Response:

As described in the Tennessee Valley Authority (TVA) response to GL 2003-01 (Reference 1, Enclosure E), the BFN control room habitability zone (CRHZ) is a significant portion of floor elevation 617' of the control building. The zone contains the following areas:

- Common Unit 1 and Unit 2 control room
- Separate Unit 3 control room
- Plant common switchyard relay equipment room
- Technical Support Center (TSC) room
- Control Room Emergency Ventilation system (CREVS)
- Equipment room
- Miscellaneous equipment rooms and office spaces on either end of the floor

The TSC is located within the CRHZ. Specifically, the TSC is located between the Unit 1/2 control room and Unit 3 control room at elevation 617. The TSC does not have a separate intake or ventilation system. Rather, the TSC ventilation air is provided by the CREVS during Design Basis Accidents (DBAs). For the DBA operator dose, no differentiation is made between the rooms within the CRHZ located on elevation 617' (e.g., Unit 1/2 control room and Unit 3 control room, the Relay Room, and the TSC). Therefore, there is not a separate dose analysis for the TSC. Rather, the TSC dose is assumed to be equal to the Control Room dose calculated for each of the respective DBAs [Loss of Coolant Accident (LOCA), Control Rod Drop Accident (CRDA), Main Steam Line Break Accident (MSLBA), and Fuel Handling Accident (FHA)].

Reference

1. Letter from NRC to TVA, "Tennessee Valley Authority – Browns Ferry Nuclear Plant (BFN) – Units 1, 2, and 3 – Dockets 50.259, -260, and -296 – Facility Operating Licenses DPR-33, -52, and -68 – Response to Generic Letter (GL) 2003-01 – Control Room Habitability," December 8, 2003 (ML033430322).

ENCLOSURE 2

ARCB-RP-RAI-01

NEDC-33860 (LAR Attachment 6 (proprietary) and Attachment 8 (non-proprietary), "Safety Analysis Report for Browns Ferry Nuclear Plant Units 1, 2, and 3 Extended Power Uprate"), Section 2.10.1.1, page 2-487, states that the installation of the new steam dryers will ensure that the moisture carryover will remain very low. What is the expected value of the moisture carryover with the new steam dryers, and how does it compare with the current design basis value?

TVA Response:

As discussed in Section 2.2.3.2.3 of NEDC-33860, the moisture carryover with the new steam dryer is expected to be ≤ 0.1 weight percent at Extended Power Uprate conditions, which is also the current value for moisture carryover.

ARCB-RP-RAI-02

NEDC-33860, Section 2.10.1.1, page 2-487, states that an increase by a factor of 1.32 in the radiation levels under extended power uprate (EPU) conditions may exceed the original design criteria of 1 millirem (mrem)/hour in the turbine building general area. Provide an estimate of the maximum dose rate above the design criteria expected in the turbine building general area under EPU conditions. Verify that the radiation zoning design basis for the facility will be updated.

TVA Response:

For the majority of the general areas in the turbine building, the design value of 1 mrem/hour will not be exceeded at Extended Power Uprate (EPU) conditions. It is noted that the 32% multiplier (which encompasses the contribution from both the additional Nitrogen-16 (N-16) as well as an increase in other sources as a result of the increased power level) should only be applied to components downstream of the main steam line piping, where residence time is longer (i.e., the turbine and condenser). For Turbine Building areas that are not directly impacted by the major steam pipes, the maximum multiplier of 20%, from Section 8.5 of the Constant Pressure Power Uprate Licensing Topical Report (CLTR), NEDC-33004P-A (Reference 1), should be used (contribution as a result of the additional carryover from the core).

As provided in Table 2.10-1c of NEDC-33860, the normal operating doses within the turbine building vary depending on location. The majority of these values are based on a measured dose rate of 1 mrem/hour in the general area, which is consistent with the original radiation design criteria of the plant. An evaluation was performed to compare these original measured dose rates to post Hydrogen Water Chemistry (HWC) implementation. The results showed that the majority of the dose rates did not change and remained less than 1 mrem/hour; however, some locations did increase by a factor of 5 with one specific location increasing by a factor of 6, which was located near the turbines. These areas are currently denoted as restricted areas (i.e., high radiation areas). Therefore, this increase does not change this designation.

Personnel exposures will be maintained within acceptable limits via the BFN-1, 2, and 3 As Low As Reasonably Achievable (ALARA)/Radiation Program. Procedural controls will compensate for increased radiation levels in the plant as a result of EPU conditions. Per the CLTR, NEDC-33004P-A (Reference 1), plants with zinc injection, HWC, and noble metal addition show a decrease in post-operation radiation levels and/or reduced repairs required in radiation areas.

The radiation shielding provided in the balance of plant (e.g., around radioactive waste systems, main steam lines, the main turbine) is conservatively designed in accordance with the GE specification for the shielding design basis to minimize the impact of the increased source terms on the occupational dose in the normally occupied areas of the plant during normal operations. Therefore, the radiation zones designations for normally occupied areas of the plant remain bounded based on the current shielding designs. For areas that have restricted access, these areas will continue to be controlled administratively to assure that personnel doses remain ALARA at EPU conditions.

Reference

1. General Electric NEDC-33004P-A, Revision 4, "Licensing Topical Report Constant Pressure Power Uprate," dated July 2003.

ARCB-RP-RAI-03

NEDC-33860, Table 2.9-12, "Post-LOCA Mission Doses," states the mission doses for the Control Room/TSC to Post-Accident Sampling Station to be 6.59 roentgen equivalent man (rem), 13.7 rem, and 14 rem for Units 1, 2 and 3, respectively. A footnote at the bottom of the table indicates that these doses, which are greater than the regulatory limit of 5 rem for a single individual, can be distributed over a number of individuals. Provide a description of the tasks involved in each of these missions and how they will be distributed such that no single individual performing these missions will exceed 5 rem.

TVA Response:

The individual tasks associated with both collecting and analyzing samples from the Post-Accident Sampling Station are presented in Table 2.9-12 of NEDC-33860. These tasks include Degassed Reactor Coolant System (RCS) Sample, Small RCS Sample, Dissolved Gas Sample, and Containment Air Sample. Each task (mission) presented in Table 2.9-12 of NEDC-33860 includes the dose received during travel to the Post-Accident Sampling Station to collect the sample as well as the dose expected during the analysis of the sample on site, if applicable.

In accordance with Table 2.9-12, the individual tasks (missions) are less than 5 rem. However, collectively, the total mission dose would exceed the regulatory limit if performed by a single individual. Therefore, the dose to an individual performing these missions would be administratively controlled by the Radiation Protection organization to ensure that no individual would exceed the regulatory limit of 5 rem. Specifically, under high dose (accident) conditions, personnel dose will be closely monitored to ensure that the individual dose does not exceed the regulatory limit. For TVA to accomplish this, the tasks (missions) may be split between two or more individuals to ensure that the regulatory limit is not exceeded. During each mission, the actual dose rates/doses obtained by radiation protection technicians taking samples will be closely monitored by radiation protection personnel at the time the samples are taken. In the event it becomes likely that the 5 Rem whole body dose limit would be exceeded, the mission would be split between two or more technicians, thereby reducing the dose received by any one technician. One individual need not perform all sampling and all analyses.

In addition, the analysis that documents the mission was previously reviewed as part of the “Browns Ferry Nuclear Plant Unit 1 Recovery – NRC Integrated Inspection Report 05000259/2006006” item E8.3 (Reference 1) with the following conclusion:

> *“The inspectors verified that a similar installation for Unit 1 is described in DCN 51185. Additionally, Calculation NDQ0043900029, “Post Accident Sampling Doses” was reviewed to determine the expected mission doses to personnel using the designed Unit 1 sampling system and the ability to meet the 5 rem whole body dose requirement. The inspectors determined that the licensee’s planned sampling system installation will meet the licensee’s commitment to be able to sample the reactor coolant, suppression pool water and containment atmosphere within a reasonable period of time for less than or equal to the doses expected from the installations on Units 2 and 3. The inspectors determined that no further*

actions were required for Unit 1. Therefore, because this modification is being tracked under the facility modification process and any deficiencies would likely be detected by the licensee's oversight programs, this item meets the closure criteria established for Unit 1 recovery issues. This issue is closed for Unit 1."

Reference

1. Letter from NRC to TVA, "Browns Ferry Nuclear Plant Unit 1 Recovery – NRC Integrated Inspection Report 05000259/2006006," dated May 15, 2006 (ML061350181).

ARCB-RP-RAI-04

NEDC-33860, page 2-494, second paragraph, and Table 2.10-2, state that the post-EPU N-16 skyshine contribution to offsite normal radiation dose is expected to be negligible. Provide justification for this conclusion.

TVA Response:

External gamma radiation levels were measured by thermoluminescent dosimeters (TLDs) or Optically-Simulated Luminescence (OSLs) deployed around the Browns Ferry Nuclear Plant (BFN) as part of the offsite Radiological Environmental Monitoring Program (REMP). The offsite doses at several locations are provided in Table 1 for the years 2009 to 2013 (Reference 1). From the table, the offsite measured doses near the site, within a 1 mile radius, are statistically higher than at several miles away from the site. The measured doses are approximately 10 to 20 mrem/year higher onsite than at offsite stations. This difference is consistent with levels measured for pre-operation and construction phases of BFN where the average radiation levels onsite were generally 8 to 24 mrem/year higher than the levels offsite. This may be attributable to natural variations in environmental radiation levels, earth moving activities onsite, the mass of concrete employed in the construction of the plants, or other undetermined influences. Fluctuations in natural background dose rates and in TLD/OSL readings tend to mask any small increments which may be due to plant operations. Thus, there is no identifiable increase in dose rate levels attributable to direct radiation from plant equipment.

Based on actual surveys of the Turbine Building roof, an estimate of the dose at the site boundary from the radiation being emitted from the Turbine Building was performed. This dose was projected to the site boundary by using a formulated skyshine dose reduction factor value of 6×10^{-7} [from Figure 4 of (Reference 2) for a distance of 1,000 meters], which results in a pre-EPU estimate of 0.42 mrem/year. Based on an increase in both N-16 and C-15 production associated with the EPU, a maximum increase in the effective radiation field from exposed piping was projected to be ~20.3% or ~0.58 mrem/year at the site boundary. Thus, the expected increase in the skyshine contribution to the site boundary dose would still not be discernable based on the historical 8 to 24 mrem/year difference seen between onsite and offsite measurements.

References

1. TVA, Radiological Environmental Monitoring Program (REMP) (2009 to 2013)

2. Technical Paper, “Hybrid Skyshine Calculations for Complex Neutron and Gamm-Ray Sources,” by J. K. Shultis, Nuclear Science and Engineering: 136, 294-304 (2000).

Table 1: Offsite Dose at Plant TLD/OSL Locations

Map Location Number	TLD Station Number	Direction degrees	Approximate Distance miles	Annual Dose for Various Years mR/yr 2009	2010	2011	2012	2013	Annual Average mR/yr
48	SE-1	130	0.5	59.2	50.6	79.1	77.3	73.6	68.0
39	NNE-2	31	0.7	61.2	61.2	82.5	77.4	70.8	70.6
41	NE-1	51	0.8	59.2	61.3	83	75.8	77.6	71.4
44	E-1	85	0.8	62.7	55.3	79.9	75.2	86.4	71.9
8	NNE-1	12	0.9	50	55	72.3	75.3	76.4	65.8
9	ENE-1	61	0.9	64.7	56.7	79.5	73.6	78	70.5
46	ESE-1	110	0.9	44.7	48	70.5	63.7	64.2	58.2
7	N-1	348	1	60.2	59.2	82.6	75.1	82.9	72.0
68	NNW-1	331	1	53.1	49.7	77.5	68.2	69.5	63.6
75	N-1A	355	1	62.2	59.9	85.9	75.7	84.7	73.7
10	NNW-2	331	1.7	57.7	56.5	75.2	64.8	72.1	65.3
55	SW-1	228	1.9	44.1	40.8	60.4	63.7	61.7	54.1
61	W-1	275	1.9	45.1	44.9	74.4	65	65.5	59.0
66	NW-1	326	2.2	33.1	31.9	57.9	52	50.5	45.1
58	WSW-1	244	2.7	36.1	31.8	55.7	52.5	50.1	45.2
47	ESE-2	112	3	46.2	41.8	71.1	62.9	60.8	56.6
53	SSW-1	203	3	35.6	36.7	57.4	48.9	55.5	46.8
51	S-1	185	3.1	47.1	42.4	71.4	62.5	63.3	57.3
64	WNW-1	291	3.3	47.2	46.7	72.7	56.7	57.1	56.1
54	SSW-2	199	4.4	44.6	40.6	60.6	63.6	57	53.3
65	WNW-2	293	4.4	44.6	40.3	63.2	57.1	57.4	52.5
56	SW-2	219	4.7	42.6	48.3	67.7	65.1	58.4	56.4

ENCLOSURE 2

Map Location Number	TLD Station Number	Direction degrees	Approximate Distance miles	Annual Dose for Various Years mR/yr 2009	2010	2011	2012	2013	Annual Average mR/yr
62	W-2	268	4.7	47.6	35	63.9	60.8	54.9	52.4
52	S-2	182	4.8	38.6	37	67.1	50	49.3	48.4
38	N-2	1	5	43.1	33.8	63	60.2	54.9	51.0
42	NE-2	49	5	52.7	50.5	77.8	74.2	68.2	64.7
50	SSE-1	163	5.1	49.7	43.3	60.5	58.3	57.7	53.9
59	WSW-2	251	5.1	46.7	46.4	68	64.5	63.4	57.8
40	NNE-3	19	5.2	41.6	42.7	66.5	59.6	55.8	53.2
45	E-2	91	5.2	42.1	42.6	68.1	58.3	60.6	54.3
69	NNW-3	339	5.2	49.2	50.5	71.7	64.6	64.2	60.0
67	NW-2	321	5.3	54.2	48.1	74	59.2	67.6	60.6
49	SE-2	135	5.4	45.2	42.3	69.8	64.4	65.5	57.4
57	SW-3	224	6	41.1	32.8				37.0
43	ENE-2	62	6.2	50.6	45.3	76.2	64.3	66.5	60.6
3	SSE-2	165	7.5	47.1	50	77.7	60.8	65.9	60.3
60	WSW-3	257	10.5	40.1	44.6	59.9	60	54.4	51.8
2	NE-3	56	10.9	42.1	44.3	67.7	56.7	65.2	55.2
1	NW-3	310	13.8	36.1	36.9	59.7	54.7	51.2	47.7
6	E-3	90	24.2	51.7	43.3	73	66	68.5	60.5
5	W-3	275	31.3	38.6	36.2	63.9	57	53.9	49.9