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Serial: RA-15-0013 10 CFR 50.90
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U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
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CATAWBA NUCLEAR STATION, UNIT NOS. 1 AND 2
DOCKET NOS. 50-413 AND 50-414
RENEWED LICENSE NOS. NPF-35 AND NPF-52

MCGUIRE NUCLEAR STATION, UNIT NOS. 1 AND 2
DOCKET NOS. 50-369 AND 50-370
RENEWED LICENSE NOS. NPF-9 AND NPF-17

OCONEE NUCLEAR STATION, UNIT NOS. 1, 2 AND 3
DOCKET NOS. 50-269, 50-270 AND 50-287
RENEWED LICENSE NOS. DPR-38, DPR-47 AND DPR-55

SUBJECT: LICENSE AMENDMENT REQUEST PROPOSING A NEW SET OF FISSION GAS GAP RELEASE FRACTIONS FOR HIGH BURNUP FUEL RODS THAT EXCEED THE LINEAR HEAT GENERATION RATE LIMIT DETAILED IN REGULATORY GUIDE 1.183, TABLE 3, FOOTNOTE 11

REFERENCE:

1. Regulatory Guide 1.183, *Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors*, Revision 0, U.S. Nuclear Regulatory Commission, July 2000.

Ladies and Gentlemen:

Pursuant to 10 CFR 50.90, Duke Energy Carolinas, LLC (Duke Energy) hereby submits a license amendment request (LAR) for Catawba Nuclear Station (CNS), Units 1 and 2; McGuire Nuclear Station (MNS), Units 1 and 2; and Oconee Nuclear Station (ONS), Units 1, 2 and 3. This request for amendment would revise the facilities as described in the Updated Final Safety Analysis Reports (UFSAR) to provide gap release fractions for high-burnup fuel rods (i.e., greater than 54 GWD/MTU) that exceed the 6.3 kW/ft linear heat generation rate (LHGR) limit detailed in Table 3 of Regulatory Guide 1.183 (Reference 1). Footnote 11 to Table 3 “Non-LOCA Fraction of Fission Product Inventory in Gap” in Reference 1 states that gap fractions calculated directly by the licensee may be considered on a "case-by-case basis." The alternative set of non-LOCA gap release fractions calculated for CNS, MNS and ONS, and submitted herein, support an increase to the Reference 1 LHGR limit.

The proposed changes in this amendment request would result in improved core designs and would minimize the number of feed assemblies, the associated rate of spent fuel pool inventory accumulation and eventual dry storage needs.

To support this license amendment request, Duke Energy provides bounding gap release fraction calculations for high-burnup fuel rods exceeding the LHGR limit. The results of the gap fraction calculations are then used to assess dose consequences for fuel-handling type accidents at CNS, MNS and ONS in which the damaged fuel assemblies include fuel rods operated beyond the Regulatory Guide 1.183, Table 3 LHGR limit in order to demonstrate that the results satisfy the acceptance criteria of both Regulatory Guide 1.183 and 10 CFR 50.67.

Enclosure 1 provides an evaluation of the proposed changes. Applicable marked-up Updated Final Safety Analysis Report (UFSAR) pages are included as Enclosure 2. The proposed amendment does not involve a change to any Operating License Condition or Technical Specification.

The proposed changes have been evaluated in accordance with 10 CFR 50.91(a)(1) using criteria in 10 CFR 50.92(c), and it has been determined that the proposed changes involve no significant hazards consideration. The bases for these determinations are included in Enclosure 1.

Staff approval of this license amendment application is requested within one year of the date of this submittal. Once approved, the license amendments will be implemented within 120 days.

There are no new regulatory commitments contained in this letter.

In accordance with 10 CFR 50.91, Duke Energy is notifying the States of North Carolina and South Carolina of this license amendment request by transmitting a copy of this letter and enclosures to the designated State Officials. Should you have any questions concerning this letter, or require additional information, please contact Art Zaremba at 980-373-2062.

I declare under penalty of perjury that the foregoing is true and correct. Executed on July 15, 2015.

Sincerely,

Regis T. Repko

Regis T. Repko, Senior Vice President
Governance, Projects and Engineering

Enclosures:

1. Evaluation of the Proposed Change
2. Proposed Updated Final Safety Analysis Report Changes (Mark-up)

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Enclosure 1

Evaluation of the Proposed Change

Subject: License Amendment Request Proposing a New Set of Fission Gas Gap Release Fractions for High Burnup Fuel Rods that Exceed the Linear Heat Generation Rate Limit Detailed in Regulatory Guide 1.183, Table 3, Footnote 11

1. SUMMARY DESCRIPTION

This evaluation supports a request to amend the Operating Licenses NPF-35, NPF-52, NPF-9, NPF-17, DPR-38, DPR-47 and DPR-55 for Catawba Nuclear Station (CNS), Unit Nos. 1 and 2, McGuire Nuclear Station (MNS), Unit Nos. 1 and 2, and Oconee Nuclear Station (ONS), Unit Nos. 1, 2 and 3.

The proposed changes would revise the facilities as described in the Updated Final Safety Analysis Report (UFSAR) to provide gap release fractions for high-burnup fuel rods (i.e., greater than 54 GWD/MTU) that exceed the 6.3 kW/ft linear heat generation rate (LHGR) limit detailed in Table 3 of Regulatory Guide 1.183 (Reference 1).

2. DETAILED DESCRIPTION

This License Amendment Request (LAR) proposes gap release fractions for high-burnup fuel rods (i.e., greater than 54 GWD/MTU) that exceed the 6.3 kW/ft LHGR limit in Footnote 11 of Table 3 in Regulatory Guide 1.183 ("Non-LOCA Fraction of Fission Product Inventory in Gap"). Footnote 11 states:

> "As an alternative *[to the non-LOCA gap fractions in Table 3 and the limits of Footnote 11]*, fission gas release calculations performed using NRC-approved methodologies may be considered on a case-by-case basis. To be acceptable, these calculations must use a projected power history that will bound the limiting projected plant-specific power history for the specific fuel load."

Duke Energy proposes to increase non-LOCA gap fractions for a maximum of 25 high-burnup fuel rods (i.e., greater than 54 GWD/MTU) in each fuel assembly that operates in the Catawba, McGuire and Oconee reactors. A detailed technical evaluation is provided in Section 3.1. The increases are as follows:

- The values in Regulatory Guide 1.183, Table 3 will be tripled for ^{85}Kr, ^{133}Xe, ^{134}Cs, and ^{137}Cs.
- The values in Regulatory Guide 1.183, Table 3 will be doubled for all other radioisotopes.

These increased gap fractions allow LHGRs up to 7.0 kW/ft for rod burnup between 54 and 60 GWD/MTU, and 6.9 kW/ft for rod burnup between 60 and 62 GWD/MTU. Future fuel cycle designs for Catawba, McGuire and Oconee may include up to 25 fuel rods per fuel assembly operated at LHGRs up to the proposed limits.

The gap release analysis performed to support the higher LHGRs is described in detail in Section 3.1. The analysis calculated specific gap fractions in accordance with the methods in the ANS 5.4 [1982] and ANS 5.4 [2011] standards (References 3 and 4). For each isotope considered, the results from the version of the standard that yielded the higher gap fraction were reported.

As input to the gap fraction calculations, the approved fuel performance codes COPERNIC (Reference 9) and PAD (Reference 10) were employed to determine nodal fuel temperatures for rod burnups from 0 to 62 GWD/MTU. The version of PAD that was used (PAD 4.0) includes a modification to allow thermal conductivity degradation modeling (described as PAD 4.0 TCD in Reference 17). The COPERNIC temperature model also accounts for thermal conductivity degradation effects. The fuel rods modeled with COPERNIC or PAD are associated with the

15x15 (Oconee) and 17x17 (Catawba and McGuire) assembly types currently operating in those reactors.

The analyses submitted herein also include evaluations of dose consequences of certain non-LOCA accidents in which the damaged fuel assemblies include 25 high-burnup fuel rods operated above 6.3 kW/ft. The accidents analyzed include only the “fuel handling” type accidents and the tornado missile accident. These are listed below in Table 1. No change in methodology is proposed for the departure from nucleate boiling (DNB) accidents (Rod Ejection and Locked Rotor), because fuel cycles for Catawba, McGuire and Oconee will be designed so that no fuel rod predicted to enter DNB will have been operated beyond the current limit in Footnote 11 for maximum LHGR. The calculations of the dose consequences of higher gap fractions for the “fuel handling” type accidents are provided in Section 3.2.

The changes proposed in this LAR would be reflected in updates to the Catawba, McGuire and Oconee Updated Final Safety Analysis Reports (UFSARs). Table 1 highlights the UFSAR sections where changes are proposed for the pertinent accidents covered by the dose analysis.

Table 1. List of Applicable Fuel Handling-Type Accidents

Site	Accident	UFSAR Sections
Catawba	Fuel Handling Accident	15.0, 15.7.4.2, 15.7.4.2.1, 15.7.4.2.2, 15.7.4.3.1, 15.7.4.3.2
	Weir Gate Drop	15.7.4.2.3, 15.7.4.3.3
	Cask Drop (Into a Fuel Cask Pit)	15.7.5
McGuire	Fuel Handling Accident	15.7.4.1, 15.7.4.2
	Weir Gate Drop	15.7.4.3
	Cask Drop (Into a Fuel Cask Pit)	15.7.4.4
	Tornado Missile Accident	15.10.3
Oconee	Fuel Handling Accident (Single Assembly Event)	15.1.10, 15.11.2.1, 15.11.2.2
	Fuel Cask Handling Accident (Multiple Assembly Event)	15.11.2.4

3. TECHNICAL EVALUATION

Gap release fractions for high-burnup rods (i.e., greater than 54 GWD/MTU) that exceed the 6.3 kW/ft LHGR limit in RG 1.183 have been calculated and are presented in Section 3.1. Gap fractions that bound the results of the gap release analysis were used to assess dose consequences for fuel handling-type accidents at McGuire, Catawba and Oconee. The dose assessments are described in Section 3.2.

3.1 Gap Release Analysis

The gap release analysis determines release fractions for a variety of volatile fission products in the gap between the pellet and cladding of a fuel rod. The computed release fractions correspond to a proposed increase in the Regulatory Guide 1.183 allowable fuel rod LHGR

above 54 GWD/MTU burnup. The results of this analysis are used as isotopic inventory input to dose calculations for fuel handling-type accidents.

Currently, McGuire, Catawba and Oconee have each implemented the Alternative Source Term (AST) method in their current licensing basis, in accordance with Regulatory Guide 1.183, for fuel handling-type accidents (References 12 through 16). Regulatory Guide 1.183 Table 3 provides gap release fractions for various volatile fission product isotopes and isotope groups, to be applied to non-LOCA accidents. This table limits the fuel rod LHGR to 6.3 kW/ft for rod burnups above 54 GWD/MTU, but a footnote to the table (Footnote 11) states that gap fractions calculated directly by the licensee may be considered on a case-by-case basis, if the calculations follow NRC-approved methodologies.

One NRC-approved method for determining gap release fractions is ANS 5.4 [1982] (Reference 3). This standard was endorsed by the NRC in their approval of Prairie Island LARs for selective implementation, and more recently, full implementation of AST (References 18 and 19).

In recent years, experimental data have demonstrated that fuel pellets undergo significant thermal conductivity degradation (TCD) at high-burnup, which increases interior fuel pellet temperatures. NRC Information Notice 2009-23 (Reference 11) discusses this issue in more detail. Higher fuel temperatures will yield larger fission gas release fractions in the ANS 5.4 [1982] model, particularly in the high-burnup range.

The ANS 5.4 [1982] standard has been revised, but the update (ANS 5.4 [2011] – see Reference 4) has not been formally endorsed by the NRC. However, the ANS 5.4 [2011] standard acknowledges the conservatism of the previous version based on additional experimental data after 1982, particularly with respect to the diffusion parameter used for the key I-131 isotope. The revised standard mandates the use of a NRC-approved fuel performance code that accounts for TCD, in determining temperature inputs for the gap fraction computations.

Because the ANS 5.4 [2011] standard is consistent with the basis for a proposed revision to Regulatory Guide 1.183 (see Reference 2), this gap release analysis considers both the ANS 5.4 [1982] and ANS 5.4 [2011] methods in a conservative manner, using fuel performance codes approved for use with McGuire, Catawba and Oconee fuel rods. For an isotope at any given burnup, the reported maximum gap fraction is taken from the ANS 5.4 standard that yields the higher result. The gap release analysis accounts for TCD and considers all long-lived and short-lived isotopes pertinent to fuel handling-type accidents.

The method employed for this analysis is described in additional detail in Section 3.1.1. Results from the specific gap fraction computations are documented in Section 3.1.3.

3.1.1 Method

ANS 5.4 [1982] provides equations to compute nodal fission gas releases for both long-lived (i.e., greater than 1-year half-life) and short-lived (i.e., less than 1-year half-life) isotopes. The standard notes the following with regard to its applicability:

> "This standard applies to radioactive noble gases (krypton and xenon) and, with lesser accuracy, to volatile fission products (iodine, cesium, and tellurium) in UO_2 and $(U,Pu)O_2$ fuel under steady-state conditions."

Background information for the development of ANS 5.4 [1982] is available in NUREG/CR-2507 (Reference 6).

ANS 5.4 [2011] provides a method for determining the release fractions of short half-life isotopes, while deferring to specific NRC-approved fuel performance codes for the calculation of release fractions for long-lived isotopes. Additional details and background information related to ANS 5.4 [2011] are provided in References 5 and 7.

The method in both versions of the ANS 5.4 standard utilizes a Booth diffusion model of the fuel, which includes empirical fits to measurement data to yield release fractions as a function of fuel temperature and burnup.

3.1.1.1 <u>Fuel Rod Types Considered</u>

The fuel rod designs listed below are considered for the fission gas release calculations. Though several other fuel designs have been used in the McGuire, Catawba and Oconee reactors, those designs are not evaluated here, as they are no longer being actively irradiated.

- MkB-HTP – This is the current 15x15 Areva fuel design being used in the Oconee reactors. The MkB-HTP (HTP) design has been employed in the Oconee reactors since 2008. Current core designs specify HTP fuel with solid 2.5 wt% U-235 axial blankets.

- MkB-HTP(Gad) – The current HTP product has been used with Gadolinia (Gd_2O_3) absorbers in the UO_2 fuel matrix for several recent Oconee core designs. The Gadolinia designs include several different concentrations, from 2.0 to 8.0 wt% Gd_2O_3.

- W-RFA – This is the current 17x17 Westinghouse fuel assembly design employed at McGuire and Catawba. The W-RFA (RFA) design uses annular, 6-inch, 2.60 wt% U-235 axial blankets at the top and bottom of the active fuel zone. This fuel type has been irradiated in the McGuire and Catawba reactors since 2000.

- W-RFA(IFBA) – The RFA fuel assemblies used in the McGuire and Catawba reactors typically include Integral Fuel Burnable Absorbers (IFBAs) on many of the fuel rods. The IFBA is a thin ZrB_2 coating on the outside of the UO_2 fuel pellet, over most of the active fuel length.

3.1.1.2 <u>Rod Operational Power Histories</u>

The core design must maintain fuel rod power peaking within limits determined from accident dose analyses. Because these rod power profiles represent allowable operational limits, they are used as the basis for a conservative set of power profiles for the gap fraction calculations. Table 2 shows the rod powers that are used in the gap release analysis for McGuire and Catawba RFA fuel and for Oconee HTP fuel. These powers bound the current core design limits. The rod powers shown are binned into time step (burnup) increments less than or equal to 2 GWD/MTU (with the exception of one HTP increment of 2.5 GWD/MTU), consistent with the restrictions of the ANS 5.4 [1982] and ANS 5.4 [2011] methods.

The nominal core average power (deposited within the fuel rod) is calculated below. The 0.974 and 0.973 values represent the fraction of total heat from fission that is deposited within the fuel rod.

<u>McGuire/Catawba:</u>

$$avg\ rod\ power = \left(\frac{3411000\ kW_{th} \times 1.02 \times 0.974}{193\ assys \times 264 \frac{rods}{assy} \times 12 \frac{ft}{rod}} \right) = 5.542\ \frac{kW}{ft}$$

<u>Oconee:</u>

$$avg\ rod\ power = \left(\frac{2568000\ kW_{th} \times 1.02 \times 0.973}{177\ assys \times 208 \frac{rods}{assy} \times 11.917 \frac{ft}{rod}} \right) = 5.809\ \frac{kW}{ft}$$

With these core average rod powers, peaking factors can be determined from the rod powers in Table 2. Note that the target rod LHGR beyond 54 GWD/MTU burnup is 7.0 kW/ft. The final two burnup steps for RFA fuel, however, are set to 6.9 kW/ft due to the desire to limit the gap release of I-131. See the computational results of the gap fraction calculations in Section 3.1.3.

Table 2. Projected Rod Powers in the Gap Release Analysis

McGuire / Catawba (RFA fuel)			Oconee (HTP fuel)	
Rod Burnup Range (GWD/MTU)	**Average Rod Power (kW/ft)**		**Rod Burnup Range (GWD/MTU)**	**Average Rod Power (kW/ft)**
0 – 2	8.757		**0 – 1**	9.237
2 – 4	8.757		**1 – 3**	9.237
4 – 6	8.757		**3 – 5**	9.237
6 – 8	8.757		**5 – 6**	9.237
8 – 10	8.757		**6 – 8**	9.237
10 – 12	8.757		**8 – 10**	9.237
12 – 14	8.757		**10 – 11**	9.237
14 – 16	8.757		**11 – 13**	9.237
16 – 18	8.757		**13 – 15**	9.237
18 – 20	8.757		**15 – 17**	9.237
20 – 22	8.702		**17 – 19**	9.004
22 – 24	8.591		**19 – 21**	9.004
24 – 26	8.591		**21 – 23**	9.004
26 – 28	8.535		**23 – 25.5**	9.004
28 – 30	8.480		**25.5 – 26**	8.540
30 – 32	8.092		**26 – 28**	8.540
32 – 34	7.815		**28 – 30**	8.540
34 – 36	7.815		**30 – 32**	8.307
36 – 38	7.649		**32 – 34**	8.307
38 – 40	7.649		**34 – 36**	8.307
40 – 42	7.649		**36 – 38**	8.307
42 – 44	7.482		**38 – 40**	8.307
44 – 46	7.482		**40 – 42**	8.249
46 – 48	7.482		**42 – 44**	8.249
48 – 50	7.482		**44 – 46**	7.959
50 – 52	7.205		**46 – 48**	7.959
52 – 54	7.205		**48 – 50**	7.000
54 – 55	7.000		**50 – 52**	7.000
55 – 56	7.000		**52 – 54**	7.000
56 – 57	7.000		**54 – 55**	7.000
57 – 58	7.000		**55 – 56**	7.000
58 – 59	7.000		**56 – 57**	7.000
59 – 60	7.000		**57 – 58**	7.000
60 – 61	6.900		**58 – 59**	7.000
61 – 62	6.900		**59 – 60**	7.000
			60 – 61	7.000*
			61 – 62	7.000*

*Though gap fractions for Oconee were calculated with 7.0 kW/ft, for simplicity the allowable rod LHGR above 60 GWD/MTU is restricted to 6.9 kW/ft for all sites.

3.1.1.3 Isotopes Considered for the Gap Release Calculations

Of the radionuclide groups discussed in Regulatory Guide 1.183, the Noble Gases, Halogens and Alkali Metals are pertinent for fuel handling-type accidents. Table 3 shows the list of isotopes, along with their Regulatory Guide 1.183 isotope category and the associated gap fraction valid for rod powers below 6.3 kW/ft when burnup exceeds 54 GWD/MTU.

Table 3. Isotopes Evaluated in the Gap Release Analysis

	Isotope	Reg Guide 1.183 Isotope Category	Reg Guide 1.183,Table 3 Gap Fraction
Long-lived (> 1-yr half-life) Isotopes	**Kr-85**	Kr-85	0.10
	Cs-134	Alkali Metals	0.12
	Cs-137	Alkali Metals	0.12
Short-lived (< 1-yr half-life) Isotopes	**I-130**	Other Halogens	0.05
	I-131	I-131	0.08
	I-132	Other Halogens	0.05
	I-133	Other Halogens	0.05
	I-134	Other Halogens	0.05
	I-135	Other Halogens	0.05
	Br-83	Other Halogens	0.05
	Br-85	Other Halogens	0.05
	Br-87	Other Halogens	0.05
	Kr-83m	Other Noble Gases	0.05
	Kr-85m	Other Noble Gases	0.05
	Kr-87	Other Noble Gases	0.05
	Kr-88	Other Noble Gases	0.05
	Kr-89	Other Noble Gases	0.05
	Xe-131m	Other Noble Gases	0.05
	Xe-133m	Other Noble Gases	0.05
	Xe-133	Other Noble Gases	0.05
	Xe-135m	Other Noble Gases	0.05
	Xe-135	Other Noble Gases	0.05
	Xe-137	Other Noble Gases	0.05
	Xe-138	Other Noble Gases	0.05
	Rb-86	Alkali Metals	0.12
	Rb-88	Alkali Metals	0.12
	Rb-89	Alkali Metals	0.12
	Rb-90	Alkali Metals	0.12
	Cs-136	Alkali Metals	0.12
	Cs-138	Alkali Metals	0.12
	Cs-139	Alkali Metals	0.12

3.1.1.4 Computation Process Using ANS 5.4 [1982]

For each of the isotopes in Table 3 (see Section 3.1.1.3), gap fractions are determined for individual axial and radial fuel nodes using different sets of equations for long-lived isotopes (which are considered stable) and short-lived isotopes. Subsections 3.1.1.4.1 through 3.1.1.4.5 discuss the computational procedures. Calculations require input nodal fuel temperatures and burnups for each time step listed in Table 2. These nodal inputs are produced by the COPERNIC or PAD fuel performance code (References 9 and 10). Note that in accordance with ANS 5.4 [1982], the diffusion coefficients determined for Kr-85 in equation (7) are multiplied by a factor of 7 for iodines and other halogens, and multiplied by a factor of 2 for cesiums and other alkali metals.

Per ANS 5.4 [1982], for each isotope considered, total gap fractions are to be calculated using the High Temperature Release equations and the Low Temperature release equations. Whichever temperature model yields a higher total gap fraction is the one reported.

3.1.1.4.1 Long-Lived Nuclides ($T_{1/2}$ > 1 YR) – High Temperature Release

The fission gas gap fraction F at the end of a single burnup increment is determined by:

$$F = 1 - g(\tau) \tag{1}$$

where:

$$\tau = D' t \tag{2}$$

In equation (2) above, t is time (in seconds), and D' (the diffusion coefficient) is defined in equation (7) below. The fractional release F_k at the end of k time steps (i.e., burnup increments) is:

$$F_k = 1 - \left\{ \sum_{i=1}^{k-1} \left[\frac{B_i(\tau_i g_i - \tau_{i+1} g_{i+1})}{D_i'} \right] + B_k \Delta t_k g_k \right\} / \sum_{i=1}^{k} B_i \Delta t_i \tag{3}$$

where: B_i is the fission product production rate during the i^{th} time step,

Δt_i is the length of the i^{th} time step; and

$$\tau_n = \sum_{i=n}^{k} D_i' \Delta t_i \tag{4}$$

for $\tau_i \leq 0.1$:

$$g_i = g(\tau_i) = 1 - 4\sqrt{\tau_i/\pi} + 3\tau_i/2 \tag{5}$$

or for $\tau_i > 0.1$:

$$g_i = g(\tau_i) = \frac{1}{15\tau_i} - \frac{6}{\tau_i}\sum_{n=1}^{3}\frac{e^{-n^2\pi^2\tau_i}}{n^4\pi^4} \quad (6)$$

$$D_i' = \left[100^{Bu_i/28000}\right] \times (D_o/a^2)e^{-Q/RT_i} \quad (7)$$

$D_o/a^2 = 0.61\ \mathrm{sec}^{-1}$

$Q = 72300\ \mathrm{cal/mol}$

$R = 1.987\ \mathrm{cal/mol\ K}$

T_i is the temperature (K) during the i^{th} time step

Bu_i is the accumulated burnup (MWD/MTU) at the midpoint of the i^{th} time step

Note that the above equations yield a gap fraction for an individual radial and axial fuel node. To determine the overall gap fraction for the entire fuel rod, the nodal gap releases must be weighted by their inventories of the isotope of interest. Appropriate inventory curves (as a function of burnup) are developed for Kr-85, Cs-134 and Cs-137 in Section 3.1.3. Nodal gap releases are further weighted volumetrically to account for any annular blanket fuel pellets.

3.1.1.4.2 Short-Lived Nuclides ($T_{1/2}$ < 1 YR) – High Temperature Release

The fission gas gap fraction F is determined using the conservative equilibrium equation:

$$F = 3\left[\frac{1}{\sqrt{\mu}}\coth\left(\sqrt{\mu}\right) - \frac{1}{\mu}\right] \quad (8)$$

where:

$$\mu = \lambda/D' \quad (9)$$

λ is the decay constant for the isotope of interest (sec^{-1}); and

D' (the diffusion coefficient) is defined in equation (7), with the following difference:

Bu_i is the accumulated burnup (MWD/MTU) at the endpoint of the i^{th} time step

Any appropriate halogen or alkali metal multiplier is also applied to this diffusion coefficient, as discussed at the beginning of Section 3.1.1.4.

As with subsection 3.1.1.4.1, the above equations yield a gap fraction for an individual radial and axial fuel node. For these short half-life isotopes, however, the overall gap fraction for the entire fuel rod is determined by weighting the nodal gap releases by the power levels of the individual nodes, along with volumetric weighting for annular blanket pellets. Any burnup dependence on short half-life isotopic inventories is ignored, as noted in item 4 of Section 3.1.2. This assumption is evaluated further in Section 3.1.3.

3.1.1.4.3 Long-Lived Nuclides ($T_{1/2}$ > 1 YR) – Low Temperature Release

The cumulative fission gas gap fraction F for the entire rod is determined using the following equation:

$$F = 7 \times 10^{-8} Bu \tag{10}$$

where Bu is the rod-average total accumulated burnup (MWD/MTU)

3.1.1.4.4 Short-Lived Nuclides ($T_{1/2}$ < 1 YR) – Low Temperature Release

The cumulative fission gas gap fraction F for the entire rod is determined using the following equation:

$$F = (1/\lambda)\left[10^{-7}\sqrt{\lambda} + 1.6 \times 10^{-12} P\right] \tag{11}$$

λ is the decay constant for the isotope of interest (sec^{-1}); and

P is the specific power for the fuel rod (MW/MTU)

3.1.1.4.5 Xenon Precursor Effects – High/Low Temperature Release

The total rod fission gas gap fraction F calculated for Xe-133 and Xe-135 in subsections 3.1.1.4.2 and 3.1.1.4.4 is adjusted for the effect of iodine precursors:

$$F_{total}^{Xe-133} = F^{I-133} + F^{Xe-133} - (F^{I-133} \times F^{Xe-133}) \tag{12}$$

$$F_{total}^{Xe-135} = F^{I-135} + F^{Xe-135} - (F^{I-135} \times F^{Xe-135}) \tag{13}$$

3.1.1.5 Computation Process Using ANS 5.4 [2011]

Gap fractions for each of the isotopes in Table 3 are determined, using either a direct result from the PAD or COPERNIC fuel performance code (for long-lived isotopes), or by computing gap releases for individual axial and radial fuel nodes (for short-lived isotopes). Subsections 3.1.1.5.1 through 3.1.1.5.3 discuss the specific procedures. Short-lived isotope calculations require input nodal fuel temperatures and burnups for each time step listed in Table 2. As with the ANS 5.4 [1982] method described in Section 3.1.1.4, these nodal inputs are produced by PAD or COPERNIC.

3.1.1.5.1 Long-Lived Nuclides ($T_{1/2}$ > 1 YR)

The long-lived isotopes listed in Table 3 (Kr-85, Cs-134 and Cs-137) are treated as stable. The Kr-85 fission gas gap fraction is taken directly from the pertinent fuel performance code, calculated at a 95/95 bounding tolerance. The fuel performance code must account for TCD in its model.

Gap fractions for Cs-134 and Cs-137 are determined by multiplying the Kr-85 release fraction by $\sqrt{2}$ in accordance with Section 5 of ANS 5.4 [2011].

3.1.1.5.2 Very Short-Lived Nuclides ($T_{1/2}$ < 6 HOURS)

The fission gas gap fraction (called the release-to-birth [R/B] ratio in this standard) for fuel radial node *i* in axial node *m*, during an irradiation period at constant temperature and power, is calculated as:

$$\left(\frac{R}{B}\right)_{i,m} = \left(\frac{S}{V}\right)_{i,m}\sqrt{\frac{\alpha_n D_{i,m}}{\lambda_n}} \tag{14}$$

where:

$$\left(\frac{S}{V}\right)_{i,m} = 120\ \mathrm{cm}^{-1} \quad \text{if } T_{i,m} \leq T_{link} \tag{15}$$

$$\left(\frac{S}{V}\right)_{i,m} = 650\ \mathrm{cm}^{-1} \quad \text{if } T_{i,m} > T_{link} \tag{16}$$

$T_{i,m}$ is the fuel temperature for radial node *i* in axial node m (K)

T_{link} is the temperature at which bubbles become interlinked on grain boundaries, per the burnup-dependent equations below:

$$T_{link} = \frac{9800}{ln(176 \times Bu_m)} + 273 \quad \text{if } Bu_m \leq 18.2\ \mathrm{GWD/MTU} \tag{17}$$

$$T_{link} = 1434 - (12.85 \times Bu_m) + 273 \quad \text{if } Bu_m > 18.2 \text{ GWD/MTU} \tag{18}$$

Bu_m is the accumulated pellet average burnup (GWD/MTU) of axial node *m*

α_n is the precursor effect with values for pertinent isotope *n* in Table 4

λ_n is the decay constant for the isotope *n* of interest (sec^{-1})

$$D_{i,m} = 7.6 \times 10^{-7} e^{-35000/T_{i,m}} + 1.41 \times 10^{-18} \dot{F}_m^{0.5} e^{-13800/T_{i,m}} + 2 \times 10^{-30} \dot{F}_m \tag{19}$$

$$\dot{F}_m = 4 \times 10^{10} LHGR_m (Diam_o^2 - Diam_i^2) \tag{20}$$

$Diam_o$ is the outer diameter of the fuel pellet (cm)

$Diam_i$ is the inner diameter of the fuel pellet (cm) [non-zero for annular pellets]

$LHGR_m$ is the local linear heat generation rate at axial node *m* (W/cm)

For equation (14), values for the precursor variable α_n are provided for specific isotopes in ANS 5.4 [2011]. Pertinent precursor coefficients are shown in Table 4. The standard also notes that if a value α_n is not listed, the precursor effect is small enough that α_n can be assumed to be unity.

The above equations yield a gap fraction for an individual radial and axial fuel node. In the same manner as described in subsection 3.1.1.4.2, the overall gap fraction for the entire fuel rod is determined by weighting the nodal gap releases by power and volume.

Table 4. Pertinent Values of α_n from ANS 5.4 [2011]

Isotope	Precursor coefficient α_n
I-132	137
I-133	1.21
I-134	4.4
Kr-85m	1.31
Kr-87	1.25
Kr-88	1.03
Kr-89	1.21
Xe-133	1.25
Xe-135m	23.5
Xe-135	1.85
Xe-137	1.07

3.1.1.5.3 Remaining Short-Lived Nuclides ($T_{1/2}$ > 6 HOURS)

The fission gas gap fraction (release-to-birth [R/B] ratio) for fuel radial node *i* in axial node *m*, is calculated as:

$$\left(\frac{R}{B}\right)_{i,m} = F_n \left(\frac{S}{V}\right)_{i,m} \sqrt{\frac{\alpha_{kr-85m} D_{i,m}}{\lambda_{kr-85m}}} \qquad (21)$$

where:

$$F_n = \left(\frac{\alpha_n \lambda_{kr-85m}}{\lambda_n \alpha_{kr-85m}}\right)^{0.25} \qquad (22)$$

In the above equation, F_n is the fractal scaling factor used for these longer-lived radioactive nuclides. Fractal scaling factors for isotopes with half-lives under 6 hours are less than ~ 1.0, with the exception of I-132. Reference 5 recommends that equation (21) be used with I-132, even though its half-life is less than 6 hours, to account for the large precursor effect of Te-132, which has a much longer half-life (3.2 days).

The diffusion coefficient in equation (21) is multiplied by a factor of 2 for any cesiums.

3.1.1.6 Computer Codes

The following computer programs were used for the calculations presented in Section 3.1.3. Each of these codes was internally validated.

- SCALE 5.1 / SAS2 – This program (Reference 8) is employed to evaluate generic isotopic inventories as a function of burnup, which are used to weight calculated nodal gap fractions.

- PAD – This is a NRC-approved fuel performance code (Reference 10) to be used with McGuire and Catawba RFA fuel. As noted in Section 2.1, the PAD version that is used (PAD 4.0) includes a modification to allow TCD modeling (designated PAD 4.0 TCD in Reference 17).

- COPERNIC – This is a NRC-approved fuel performance code (Reference 9) to be used with Oconee HTP fuel.

- gapfrac – This is a Visual Basic for Applications (VBA) program that computes fission gas gap fractions in accordance with ANS 5.4 [1982] and ANS 5.4 [2011], using the methods described in Sections 3.1.1.4 and 3.1.1.5. This code requires PAD or COPERNIC output rod power histories, along with nodal fuel temperatures and burnups.

3.1.2 Assumptions / Calculation Bases

The following assumptions and bases were employed for the gap release analysis:

1. Nominal (i.e., best estimate) fuel rod design/operational input was used for the PAD and COPERNIC models.

2. The McGuire, Catawba and Oconee rod operational power histories selected for this analysis (see Table 2) bound the limiting plant-specific power histories, in accordance with Footnote 11 to Table 3 of Regulatory Guide 1.183.

3. The Regulatory Guide 1.183 Fuel Rod LHGR limit above 54 GWD/MTU burnup (i.e., 6.3 kW/ft) is associated with the heat produced in the fuel (~ 0.973 fraction of total power produced), and does not include energy deposited directly to the coolant.

4. It is sufficient to characterize the inventories of short half-life isotopes (e.g., I-131) as dependent only on instantaneous power level. Any burnup-dependent effects were deemed negligible or otherwise dispositioned. Section 3.1.3 evaluates this assumption.

5. For each of the McGuire, Catawba and Oconee reactors, 102% of nominal original reactor power was used as the “baseline” operating power in PAD and COPERNIC. This bounds the Measurement Uncertainty Recapture power uprates that have been or will be implemented at the sites. The 102% power corresponds to 3479 MW_{th} for McGuire and Catawba, and 2619 MW_{th} for Oconee.

6. For sufficient detail in the gap fraction calculations, all fuel rod evaluations were performed using 24 equally-spaced axial fuel segments and 10 (PAD) or 15 (COPERNIC) equal-volume radial rings in the fuel pellet. The ANS 5.4 [1982] standard requires at least 6 radial nodes of equal volume, while the ANS 5.4 [2011] standard requires at least 7 equal-volume radial nodes. Both standards require 10 or more axial nodes of equal length for the gap fraction computations.

7. Fuel assembly axial burnup and power data from recent core designs were employed to determine appropriate axial power shapes for the fuel performance codes.

8. Steady state reactor power operation was assumed for applicability to fuel handling accidents. No major transients are considered that could release significant quantities of volatile fission products to the fuel rod gap.

9. For each of the isotopes considered, the highest gap fraction was taken from variations on central fuel enrichment, presence or absence of integral poisons and gas release computational method (ANS 5.4 [1982] versus ANS 5.4 [2011]).

10. The TCD model in the PAD code is assumed to be valid, even though the model has not been reviewed by the NRC for the current licensed version of PAD. In Section 3.1.3, gapfrac results from the TCD cases with fuel temperatures generated by PAD were compared with those generated by COPERNIC to verify the adequacy of this assumption.

3.1.3 Fission Gas Release Analysis – Calculations/Results

Section 3.1.1 described the method that is used to compute gap fractions for RFA and HTP fuel rods with the power history profiles shown in Table 2. The computer programs used for the calculations were discussed in Section 3.1.1.6.

The first step in the actual analysis was to build input decks for the PAD and COPERNIC codes so that appropriate nodal fuel temperatures could be obtained for input into the ANS 5.4 [1982] and ANS 5.4 [2011] gap release equations.

Input information for the PAD and COPERNIC cases includes:

- Fuel rod dimensions and mechanical design data
- Fuel rod backfill pressures
- Number of axial nodes modeled
- Axial power shape information
- Number of burnup time steps
- Rod power history
- Enrichments and axial blanket details
- Reactor core operational data

It is important to use axial power shapes that are accurate, especially in the high-burnup range that is of particular concern (54 to 62 GWD/MTU). Axial shapes were obtained from recent McGuire, Catawba and Oconee core designs, for fuel assemblies at the end of full power operation that had both high burnup and relatively high power. Average axial burnup shapes from these candidate fuel assemblies were used to "burn in" appropriate fuel rod nodal burnups through 54 GWD/MTU in PAD or COPERNIC. Then the average axial power shapes from these fuel assemblies were used to more accurately determine fuel performance code temperatures for rod burnups above 54 GWD/MTU.

Figures 1 and 2 show the averaged burnup and power profiles obtained from the McGuire, Catawba and Oconee core designs. The axial burnup profiles in Figure 1 are remarkably similar for all sites. The axial power shapes for McGuire and Catawba in Figure 2 are nearly the same, but the Oconee power shape is skewed somewhat toward the top of the active fuel. This is attributable to the routine withdrawal of axial power shaping rods from the Oconee cores near end-of-cycle.

As the McGuire and Catawba burnup and power shapes are nearly the same, the McGuire shapes were used in the PAD input for all RFA fuel rod cases. The Oconee burnup and power shapes were used in the COPERNIC input for all HTP fuel rod cases.

With the pertinent axial shapes from Figures 1 and 2, several PAD and COPERNIC cases were run, with variations on central fuel enrichment and presence of integral poisons. Additionally, to assess the performance of PAD and COPERNIC against the NRC FRAPCON code, cases were constructed to evaluate the bounding power profile for PWR History #6 in Reference 5.

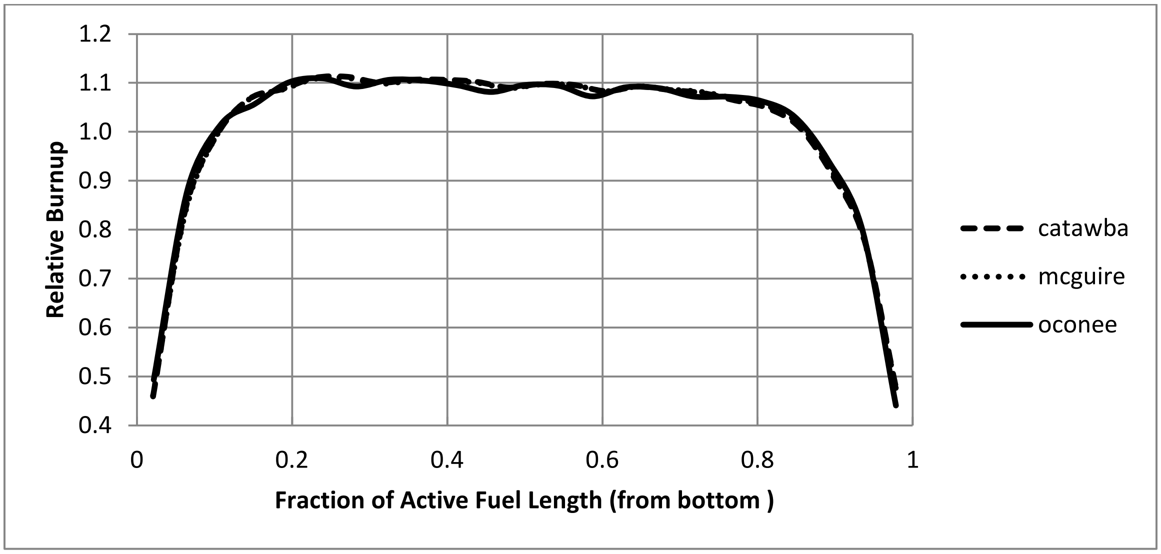

Figure 1. Average Axial Burnup Shapes for Candidate Fuel Assemblies

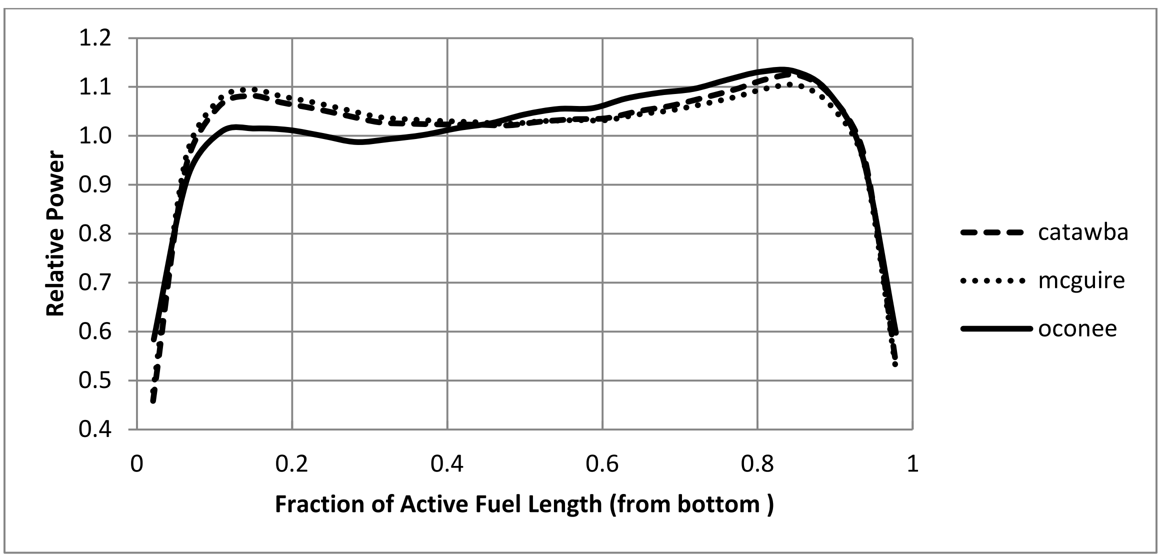

Figure 2. Average Axial Power Shapes for Candidate Fuel Assemblies

The results from the PAD and COPERNIC cases show that, regardless of fuel type, fission gas release is maximized for fuel rods with 5.0 wt% U-235 central enrichment, and without any integral poisons.

Two additional steps were required to be performed prior to calculating gap fractions with the methods outlined in Sections 3.1.1.4 and 3.1.1.5. First, fission product inventories as a function of burnup had to be determined, so that proper weighting of nodal gap fractions could be carried out. To do this, SCALE 5.1 / SAS2 computations were performed, irradiating RFA fuel and HTP fuel (at a constant nominal power) to a burnup of 100 GWD/MTU. This high burnup value was chosen because the radial nodes of fuel at the outer edge of the fuel pellet can experience such burnups near end of life.

The relative inventories of both long-lived and short-lived isotopes are very similar in comparing the results of these SCALE cases.

For most of the isotopes listed in Table 3, the inventory is the result of the fission yield (primarily from U-235, Pu-239 or Pu-241 fissions) of the isotope and its chain of radioactive precursors. However, for four of these isotopes (Cs-134, Cs-136, I-130 and Rb-86) there is very little fission yield because their would-be precursors (Xe-134, Xe-136, Te-130 and Kr-86) are stable. For these four isotopes, inventory builds up slowly with burnup due to neutron capture of Cs-133, Cs-135, I-129 and Rb-85.

Figures 3 and 4 show the inventories of important long-lived and short-lived isotopes, normalized to end-of-life values. Note that there are significant non-linearities in the Kr-85 and Cs-134 inventories in Figure 3. To account for this in the gap release calculations, polynomial curve fits were made to raw inventory values from one of the SCALE cases discussed above. The coefficients from the curve fits are shown below, where *x* is burnup (expressed in terawatt-days/MTU):

$$Kr85 = 0.000414 + 5.593x - 43.58x^2 + 147.918x^3$$

$$Cs134 = 1.2376x + 369.849x^2 - 1986.09x^3$$

$$Cs137 = 127.609x - 110.416x^2$$

The gap release results provided later in this section show that the concave shape of the Cs-134 inventory buildup yields a lower gap fraction than the nearly-linear shape of the Cs-137 buildup (see the results in Figures 5 through 7). In light of this fact, it is conservative to ignore the effect of the similar concave shapes that would be observed for the short-lived isotopes Cs-136, I-130 and Rb-86, whose inventories accumulate in the same manner.

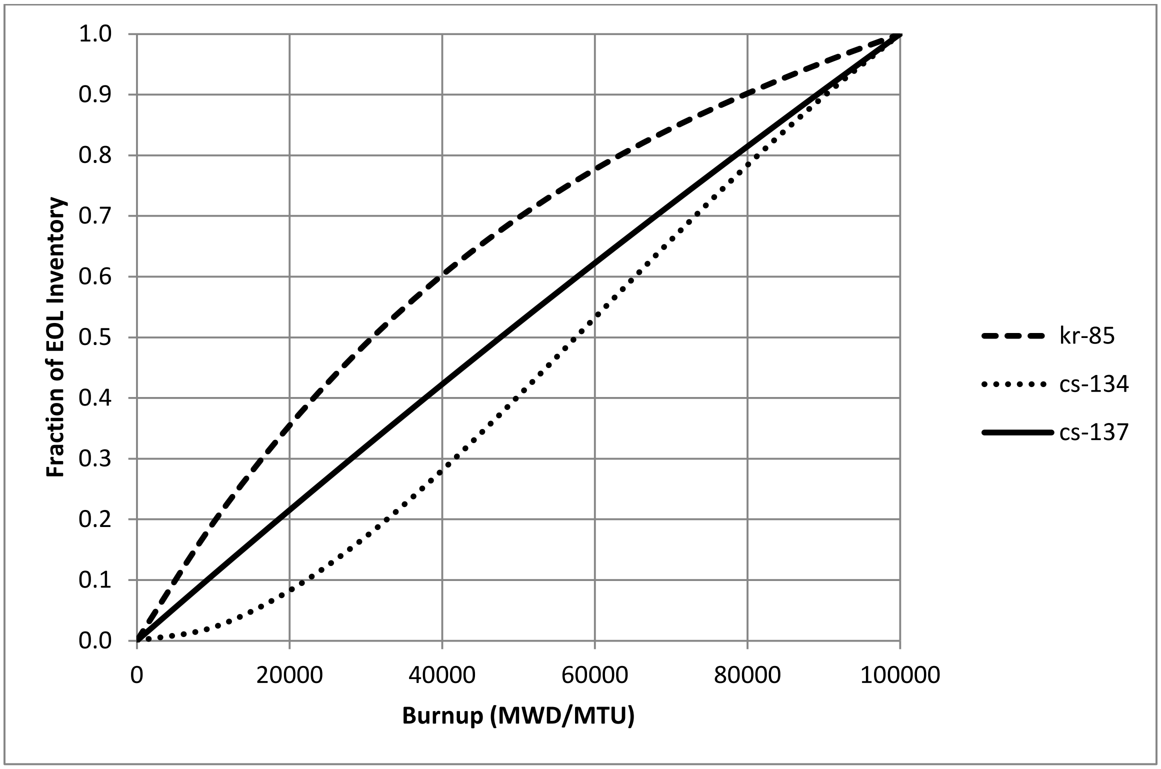

Figure 3. EOL-Normalized Inventory of Long-lived Isotopes as a Function of Fuel Burnup

Figure 4 shows only small relative inventory variations over the life of the fuel for other important short-lived isotopes. Based on these results, it is judged acceptable to compute the inventory of these isotopes (as well as the less-important short-lived isotopes in Table 3) as a direct relation to instantaneous power level. This is consistent with the nodal gap fraction weighting calculation for short-lived isotopes in ANS 5.4 [2011].

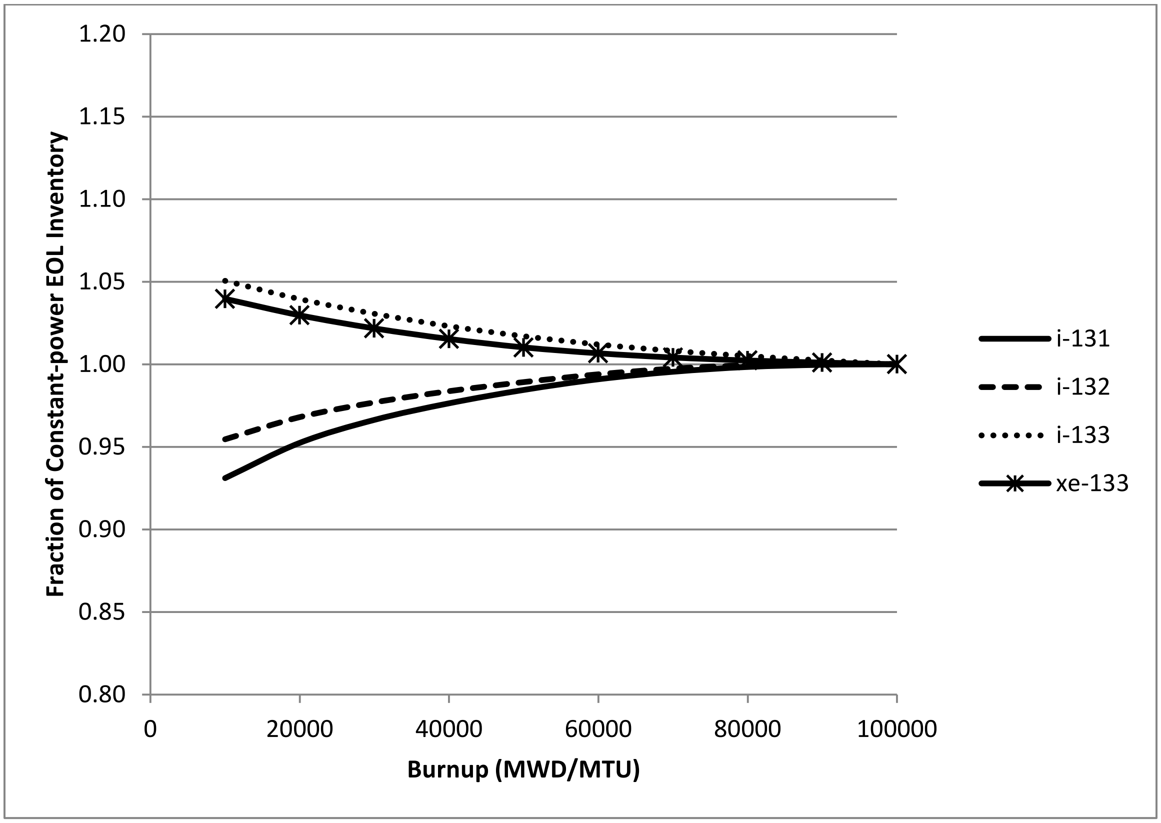

Figure 4. EOL-Normalized Constant-Power Inventory of Important Short-lived Isotopes as a Function of Fuel Burnup

With these preliminary steps completed, the gap fractions for the Table 3 isotopes were calculated, using the PAD and COPERNIC temperatures and power histories from the limiting cases. To perform the ANS 5.4 [1982] and ANS 5.4 [2011] gap fraction calculations, the gapfrac Visual Basic for Applications (VBA) program was written. This program applies the methods outlined in Section 3.1.1 to determine isotope gap release fractions for the entire fuel rod irradiation history.

The ANS 5.4 [1982] "high temperature" results from gapfrac runs for the limiting RFA and HTP cases are shown in Figures 5 and 6, for the most important long-lived and short-lived isotopes. Note the dips in the gap fractions for short-lived isotopes as power levels are reduced, and these isotopes quickly establish lower equilibrium concentrations in the gap.

Figure 7 is an example print of part of the ANS 5.4 [1982] "high temperature" results spreadsheet produced by gapfrac for RFA fuel, which was input data for Figure 5. Note that the "Rod Avg" powers shown in this figure need to be multiplied by the 0.974 factor described in Section 3.1.1.2 to obtain the LHGR for the fuel rod.

The gapfrac results from the RFA and HTP cases show that ANS 5.4 [1982] "high temperature" model (described in Section 3.1.1.4) gives much higher gap fractions than the ANS 5.4 [1982] "low temperature" model, or the calculations for short-lived isotopes in ANS 5.4 [2011].

Table 5 lists the maximum gap fractions calculated for each isotope from Table 3, for both RFA and HTP fuel. In addition to these results, it is important to also consider the directly-computed maximum PAD or COPERNIC gap release for Kr-85. As noted in subsection 3.1.1.5.1, in accordance with ANS 5.4 [2011], gap fractions for Cs-134 and Cs-137 are determined by multiplying the Kr-85 release fraction by $\sqrt{2}$. Doing this gives a maximum Cs-134/Cs-137 gap fraction of 0.285, which is about 2.4 times the Regulatory Guide 1.183 Table 3 value for alkali metals.

From the above discussion, as well as the Table 5 results, it is clear that, if it is desired to exceed a 6.3 kW/ft LHGR above 54 GWD/MTU, increased gap fractions must be accounted for in dose analyses. The results of this analysis demonstrate that with the chosen power histories in Table 2, calculated gap fractions remain below 3 times the Regulatory Guide 1.183 Table 3 values for Kr-85, Cs-134, Cs-137, and Xe-133. Gap fractions remain below 2 times the Regulatory Guide 1.183 Table 3 values for all other isotopes in Table 3.

As an additional check on the validity of the computed fuel temperatures and gap releases, comparison cases were run, modeling the Reference 5 bounding PWR power profile for RFA and HTP fuel rods. The PAD calculations for RFA fuel were performed both with and without TCD. The COPERNIC calculations for HTP fuel inherently account for TCD effects. The gapfrac code computed gap fractions using the output fuel temperatures from these PAD and COPERNIC cases.

The results from these cases are all plotted together in Figure 8. Note the general good agreement of the HTP and RFA fuel types for this Reference 5 bounding PWR power profile, when TCD is included. The TCD gap fractions shown in Figure 8 also compare quite favorably with the FRAPCON results for this power profile, taken from Figure 2.5 of Reference 5. The plot of RFA gap fractions without TCD illustrates the large effect of conductivity degradation on fuel temperatures and gap releases, especially toward end-of-life.

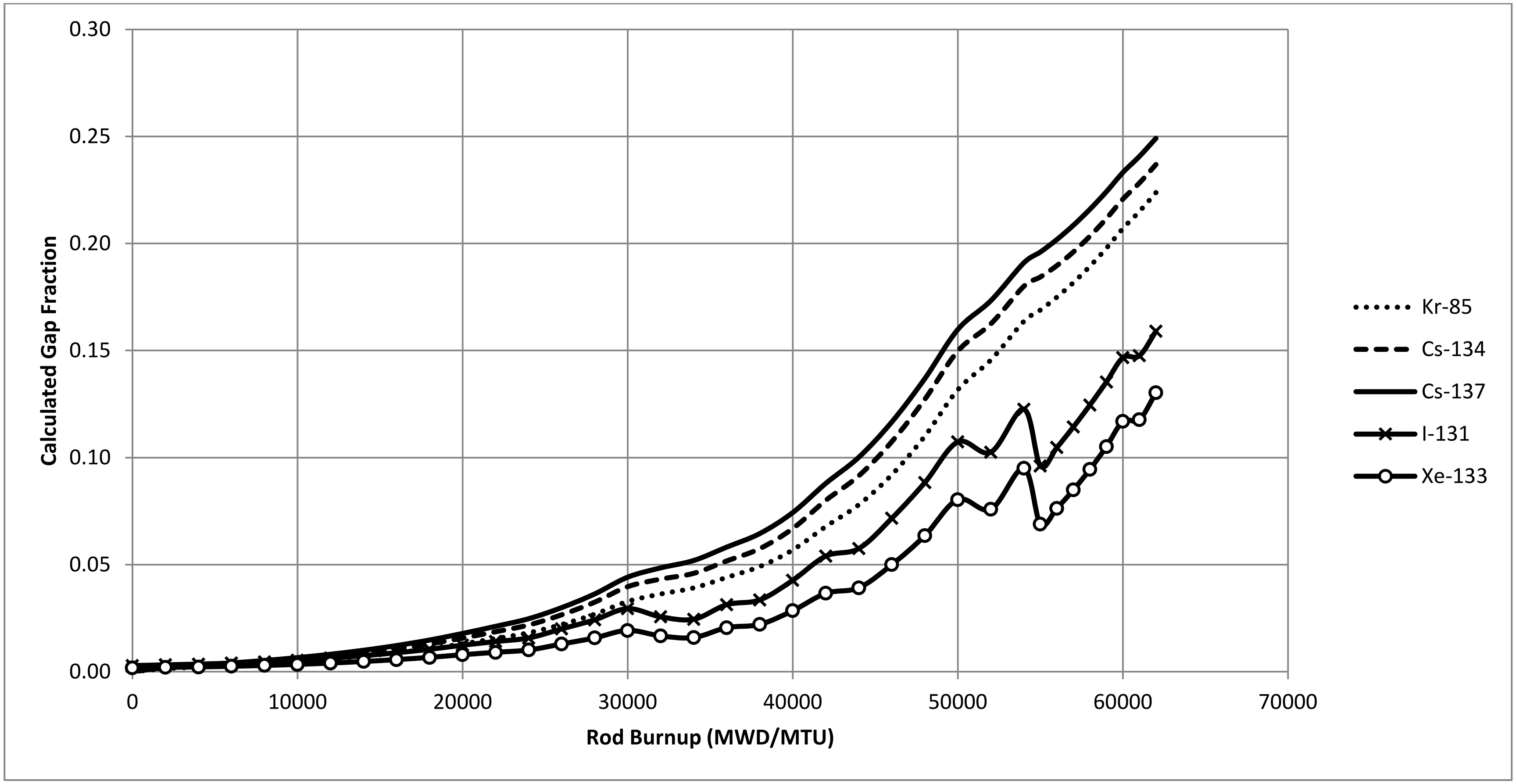

Figure 5. Calculated Gap Fractions for Important Isotopes – 5.0 wt % U-235 RFA Fuel (non-IFBA)

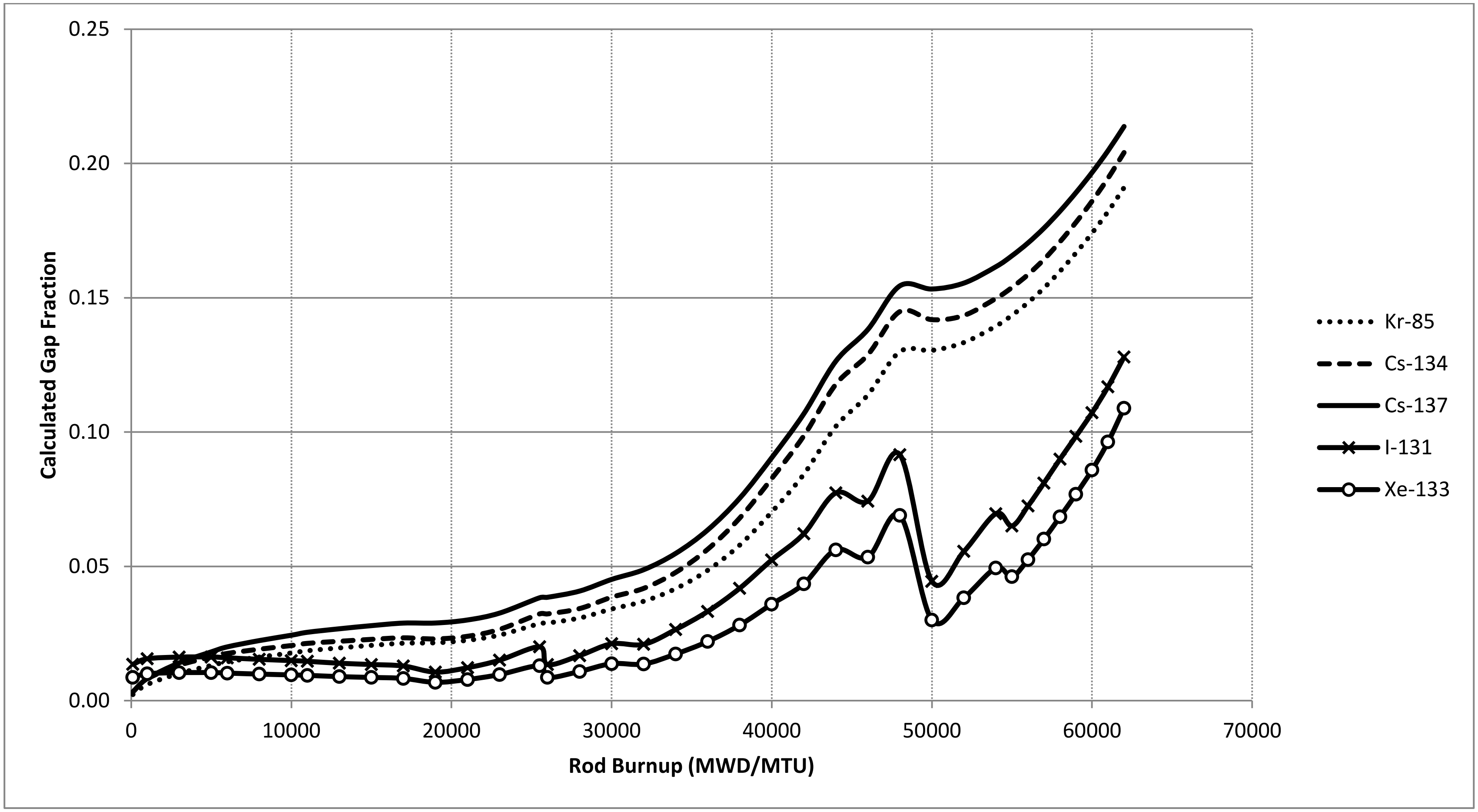

Figure 6. Calculated Gap Fractions for Important Isotopes – 5.0 wt % U-235 HTP Fuel (non-gadolinia)

High-temp Gas Release			Long-lived (>1-yr half-life)			Short-lived (<1-yr half-life) Isotopes			
1982 ANS 5.4			Isotopes						
	"Rod Avg"	Rod Total							
Time	Total Power	Burnup							
Step	(kw/ft)	(MWD/MTU)	Kr-85	Cs-134	Cs-137	I-130	I-131	I-132	I-133
1	0.860	0	0.00001	0.00001	0.00001	0.00073	0.00288	0.00031	0.00095
2	8.992	2000	0.00143	0.00197	0.00198	0.00082	0.00322	0.00035	0.00106
3	8.992	4000	0.00220	0.00280	0.00306	0.00091	0.00357	0.00039	0.00118
4	8.992	6000	0.00295	0.00364	0.00411	0.00102	0.00402	0.00044	0.00133
5	8.992	8000	0.00378	0.00461	0.00526	0.00118	0.00463	0.00051	0.00153
6	8.992	10000	0.00475	0.00575	0.00660	0.00138	0.00541	0.00059	0.00179
7	8.991	12000	0.00590	0.00713	0.00818	0.00163	0.00638	0.00070	0.00211
8	8.992	14000	0.00726	0.00876	0.01004	0.00193	0.00752	0.00083	0.00249
9	8.992	16000	0.00887	0.01070	0.01223	0.00228	0.00888	0.00098	0.00295
10	8.992	18000	0.01077	0.01300	0.01482	0.00270	0.01049	0.00116	0.00349
11	8.992	20000	0.01304	0.01574	0.01788	0.00320	0.01240	0.00137	0.00413
12	8.935	22000	0.01552	0.01869	0.02121	0.00365	0.01410	0.00157	0.00471
13	8.821	24000	0.01819	0.02184	0.02477	0.00409	0.01577	0.00176	0.00528
14	8.821	26000	0.02207	0.02661	0.02995	0.00522	0.02001	0.00225	0.00674
15	8.764	28000	0.02687	0.03249	0.03634	0.00637	0.02425	0.00275	0.00822
16	8.707	30000	0.03283	0.03973	0.04418	0.00780	0.02946	0.00337	0.01005
17	8.309	32000	0.03631	0.04327	0.04852	0.00675	0.02565	0.00291	0.00871
18	8.024	34000	0.03915	0.04597	0.05196	0.00644	0.02453	0.00278	0.00831
19	8.024	36000	0.04407	0.05175	0.05822	0.00831	0.03133	0.00360	0.01072
20	7.853	38000	0.04913	0.05750	0.06453	0.00893	0.03355	0.00387	0.01151
21	7.853	40000	0.05690	0.06690	0.07435	0.01155	0.04273	0.00501	0.01487
22	7.853	42000	0.06782	0.08004	0.08797	0.01490	0.05401	0.00649	0.01914
23	7.683	44000	0.07801	0.09153	0.10017	0.01597	0.05753	0.00696	0.02050
24	7.683	46000	0.09203	0.10748	0.11667	0.02046	0.07169	0.00897	0.02621
25	7.683	48000	0.11003	0.12725	0.13692	0.02618	0.08834	0.01154	0.03341
26	7.683	50000	0.13173	0.14979	0.15986	0.03339	0.10734	0.01484	0.04242
27	7.398	52000	0.14562	0.16248	0.17331	0.03143	0.10257	0.01394	0.03999
28	7.398	54000	0.16363	0.18004	0.19099	0.03973	0.12263	0.01779	0.05028
29	7.189	55000	0.16894	0.18437	0.19603	0.02842	0.09608	0.01253	0.03628
30	7.189	56000	0.17494	0.18970	0.20186	0.03159	0.10474	0.01397	0.04025
31	7.189	57000	0.18175	0.19607	0.20852	0.03529	0.11435	0.01567	0.04487
32	7.189	58000	0.18938	0.20342	0.21598	0.03944	0.12456	0.01759	0.05003
33	7.189	59000	0.19782	0.21167	0.22422	0.04412	0.13536	0.01978	0.05581
34	7.189	60000	0.20704	0.22076	0.23318	0.04936	0.14670	0.02226	0.06223
35	7.086	61000	0.21497	0.22828	0.24069	0.04971	0.14758	0.02242	0.06266
36	7.086	62000	0.22377	0.23688	0.24911	0.05540	0.15902	0.02515	0.06958

Figure 7. Example gapfrac Output (5.0 wt % U-235 RFA Fuel)

Table 5. Bounding Results from gapfrac Calculations

Isotope	Isotope Category	Reg Guide 1.183 Table 3 Value	McGuire/Catawba	Oconee	Max Ratio
Long-lived (> 1-yr half-life) Isotopes					
Kr-85	Kr-85	0.10	0.224	0.191	**2.24**
Cs-134	Alkali Metals	0.12	0.237	0.204	1.97
Cs-137	Alkali Metals	0.12	0.249	0.214	2.08
Short-lived (< 1-yr half-life) Isotopes					
I-130	Other Halogens	0.05	0.055	0.046	1.11
I-131	I-131	0.08	0.159	0.128	**1.99**
I-132	Other Halogens	0.05	0.025	0.021	0.50
I-133	Other Halogens	0.05	0.070	0.058	1.39
I-134	Other Halogens	0.05	0.016	0.014	0.32
I-135	Other Halogens	0.05	0.041	0.035	0.83
Br-83	Other Halogens	0.05	0.026	0.022	0.52
Br-85	Other Halogens	0.05	0.004	0.003	0.08
Br-87	Other Halogens	0.05	0.002	0.002	0.04
Kr-83m	Other Nobles	0.05	0.009	0.008	0.18
Kr-85m	Other Nobles	0.05	0.014	0.012	0.27
Kr-87	Other Nobles	0.05	0.007	0.006	0.15
Kr-88	Other Nobles	0.05	0.011	0.009	0.22
Kr-89	Other Nobles	0.05	0.001	0.001	0.03
Xe-131m	Other Nobles	0.05	0.092	0.076	1.84
Xe-133m	Other Nobles	0.05	0.044	0.037	0.88
Xe-133	Other Nobles	0.05	0.130	0.109	**2.61**
Xe-135m	Other Nobles	0.05	0.003	0.003	0.07
Xe-135	Other Nobles	0.05	0.060	0.051	1.20
Xe-137	Other Nobles	0.05	0.002	0.001	0.03
Xe-138	Other Nobles	0.05	0.003	0.003	0.06
Rb-86	Alkali Metals	0.12	0.140	0.113	1.16
Rb-88	Alkali Metals	0.12	0.005	0.004	0.04
Rb-89	Alkali Metals	0.12	0.005	0.004	0.04
Rb-90	Alkali Metals	0.12	0.002	0.002	0.02
Cs-136	Alkali Metals	0.12	0.124	0.101	1.04
Cs-138	Alkali Metals	0.12	0.007	0.006	0.06
Cs-139	Alkali Metals	0.12	0.004	0.003	0.03

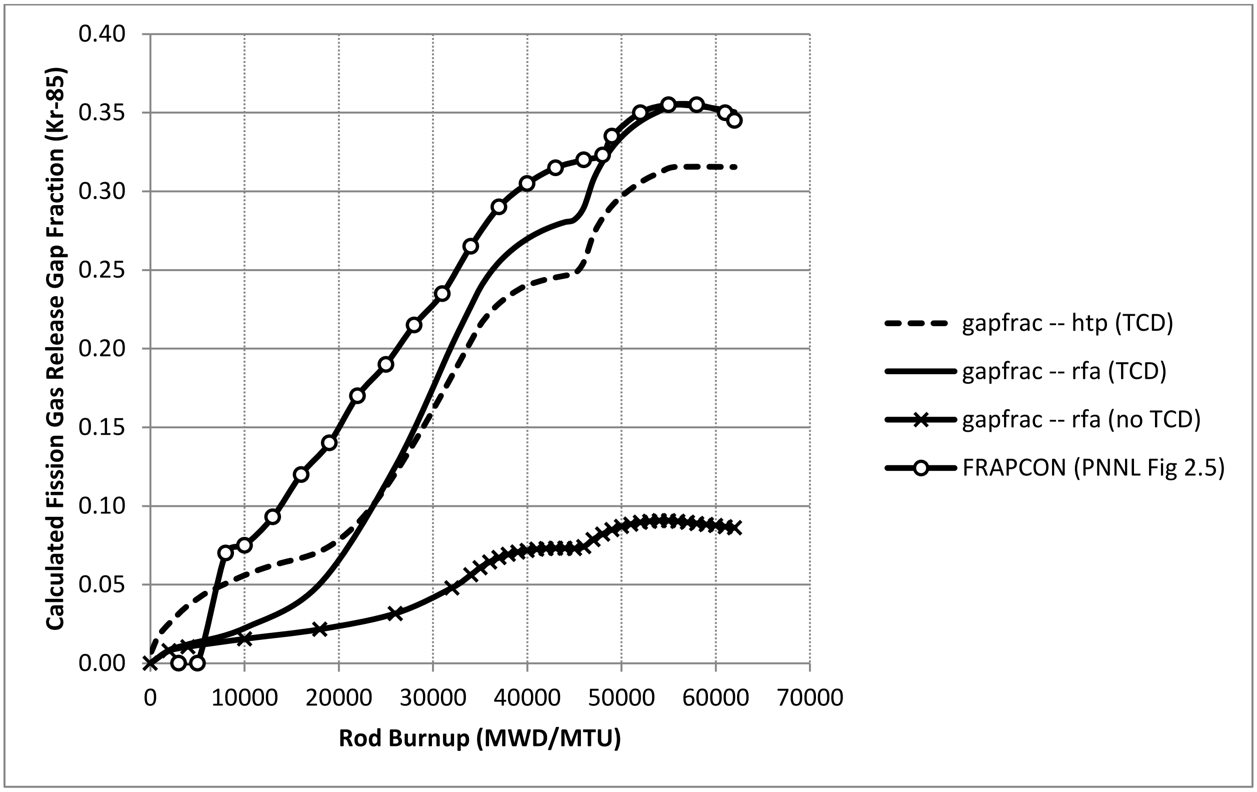

Figure 8. gapfrac (HTP and RFA Fuel) and FRAPCON (from Reference 5) Kr-85 Gap Fractions for PNNL Bounding Power History – gapfrac results from ANS 5.4 [1982] "high temperature" calculations

3.1.4 Conclusions

With bounding operational power histories applied to the fuel rods in each of the McGuire, Catawba, and Oconee reactors for burnups up to 54.0 GWD/MTU, maximum linear heat generation rates that exceed the 6.3 kW/ft limit in Regulatory Guide 1.183 have been evaluated for the remainder of the allowable rod burnup:

- 54.0 to 60.0 GWD/MTU – 7.0 kW/ft
- 60.0 to 62.0 GWD/MTU – 6.9 kW/ft

The conservative rod power histories in Table 2 have been analyzed, using NRC-approved fuel performance codes (PAD or COPERNIC), to obtain a fine mesh of fuel temperatures that include the effects of thermal conductivity degradation at high burnup. With these fuel temperatures, fission gas release calculations have been performed in conformance with the methods described in the ANS 5.4 [1982] and ANS 5.4 [2011] standards.

The calculations in Section 3.1.3 show that, for the isotopes considered in fuel handling accident dose analyses, the Regulatory Guide 1.183 Table 3 gap fractions must be increased. Computed fission gas release gap fractions for all reactors remain below the bounding values shown in Table 6.

Table 6. Bounding Increased Gap Fractions for Application to Fuel Handling Accidents at McGuire, Catawba, and Oconee

Isotope or Isotope Group	Gap Fraction from Table 3 of Reg Guide 1.183 (Rev. 0)	Bounding Gap Fraction from Section 3.1.3	Ratio
I-131	0.08	0.16	2
Kr-85	0.10	0.30	3
Xe-133 (other Noble)	0.05	0.15	3
Other Noble Gases	0.05	0.10	2
Other Halogens	0.05	0.10	2
Cs-134 (Alkali Metal)	0.12	0.36	3
Cs-137 (Alkali Metal)	0.12	0.36	3
Other Alkali Metals	0.12	0.24	2

3.2 Dose Consequences

With the gap release analysis complete, the dose consequences for each applicable accident listed in Table 1 were calculated. It was assumed that no more than 25 fuel rods per assembly could exceed the maximum linear heat generation rate limit of 6.3 kW/ft for burnups exceeding 54 GWD/MTU.

The source term used for Catawba and McGuire accidents in this LAR accounts for assembly operation with a constant peaking factor of 1.65. The source term used for Oconee accidents in the LAR is unchanged from the current licensing basis source term. Table 7 provides a listing of the source terms used for Catawba, McGuire and Oconee, before gap fractions and any applicable decay times are applied.

Based on the results provided in Table 6 (see Section 3.1), the gap release fractions for rods that exceed the 6.3 kW/ft LHGR limit above 54 GWD/MTU would be doubled, except for four isotopes: ^{85}Kr, ^{133}Xe, ^{134}Cs, and ^{137}Cs. The release fractions for these four isotopes would be tripled. Table 8 reiterates the increased gap fraction values determined in Section 3.1.

The fuel handling-type accidents were evaluated for Catawba, McGuire, and Oconee. Table 9 shows the dose consequences due to an increase in the gap release fractions for these 25 rods per assembly. The revised doses satisfy the requirements set forth in Regulatory Guide 1.183 and 10 CFR 50.67. Note that the Catawba UFSAR does not report radiation doses at the boundary of the Low Population Zone (LPZ) for these accidents, because the Exclusion Area Boundary (EAB) is bounding.

Table 7. Single Assembly Source Term for Catawba, McGuire, and Oconee

Isotope	Catawba/McGuire Source Term (Ci)	Oconee Source Term (Ci)
I-130	3.95E+04	3.21E+04
I-131	8.09E+05	6.82E+05
I-132	1.18E+06	9.83E+05
I-133	1.67E+06	1.38E+06
I-134	1.95E+06	1.62E+06
I-135	1.60E+06	1.31E+06
Kr-83m	1.32E+05	1.10E+05
Kr-85m	2.98E+05	2.53E+05
Kr-85	7.48E+03	8.53E+03
Kr-87	6.15E+05	5.18E+05
Kr-88	8.69E+05	7.07E+05
Kr-89	1.12E+06	9.02E+05
Xe-131m	1.24E+04	1.02E+04
Xe-133m	5.20E+04	4.27E+04
Xe-133	1.65E+06	1.31E+06
Xe-135m	3.62E+05	3.02E+05
Xe-135	4.12E+05	4.26E+05
Xe-137	1.55E+06	1.27E+06
Xe-138	1.59E+06	1.30E+06
Rb-86	2.54E+03	2.30E+03
Rb-88	8.89E+05	7.11E+05
Rb-89	1.18E+06	9.43E+05
Rb-90	1.12E+06	8.89E+05
Cs-134	2.06E+05	2.13E+05
Cs-136	5.92E+04	6.05E+04
Cs-137	9.23E+04	9.71E+04
Cs-138	1.66E+06	1.38E+06
Cs-139	1.58E+06	1.31E+06
Br-83	1.31E+05	1.10E+05
Br-85	2.99E+05	2.53E+05
Br-87	4.95E+05	4.15E+05

Table 8. Gap Release Fractions for Rods that Exceed 6.3 kW/ft Above 54 GWD/MTU

Group	Fraction
I-131	0.16
Kr-85	0.30
Xe-133	0.15
Cs-134	0.36
Cs-137	0.36
Other Noble Gases	0.10
Other Halogens	0.10
Alkali Metals	0.24

Table 9. Accident Dose Consequences

Accident	Baseline Doses (Rem TEDE)			Revised Doses (Rem TEDE)		
	EAB	LPZ	Control Room	EAB	LPZ	Control Room
Catawba Nuclear Station (see Note)						
Fuel Handling Accident	1.59	NA	2.37	**1.76**	**NA**	**2.59**
Weir Gate Drop	2.44	NA	3.87	**2.68**	**NA**	**4.24**
Cask Drop	0.005	0.0007	0.001	**0.006**	**0.0008**	**0.001**
McGuire Nuclear Station						
Fuel Handling Accident	2.95	0.27	3.52	**3.25**	**0.29**	**3.86**
Weir Gate Drop	5.60	0.51	2.97	**6.16**	**0.56**	**3.25**
Cask Drop	0.009	0.0008	0.0005	**0.01**	**0.0009**	**0.0006**
Tornado Missile Accident	2.58	2.59	4.25	**2.84**	**2.86**	**4.66**
Oconee Nuclear Station						
Fuel Handling Accident (Single Assembly Event)	1.18	0.13	2.19	**1.33**	**0.14**	**2.45**
Fuel Cask Handling Accident (Multiple Assembly Event)	1.83	0.19	3.61	**2.05**	**0.22**	**4.05**

Note: The baseline analyses for Catawba account for fuel assembly isotopic activities calculated by setting the radial peaking factor to 1.65. The radiation doses reported in the Catawba Nuclear Station UFSAR for the fuel handling accident, weir gate drop, and cask drop are associated with the radial peaking factor set to burnup dependent values.

4. REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

10 CFR 50.67 / Regulatory Guide 1.183

Regulatory Guide (RG) 1.183 provides an Alternative Source Term (AST) that is acceptable to the NRC Staff. Following the guidance in RG 1.183, Duke Energy adopted an AST that was approved by the NRC staff for use in the design basis radiological consequence analyses at ONS, MNS and CNS. Fundamental to the definition of an AST according to RG 1.183 are gap release fractions, and Table 3 of the RG provides gap release fractions for various volatile fission product isotopes and isotope groups, to be applied to non-Loss of Coolant Accident (LOCA) accidents. The release fractions are valid only if the maximum LHGR does not exceed the RG 1.183 value of 6.3 kW/ft for rod burnup above 54 GWD/MTU. In order to exceed the RG 1.183 maximum LHGR above 54 GWD/MTU, increased gap release fractions must be determined and accounted for in the dose analyses. Increased gap release fractions were determined by Duke Energy and were accounted for in the ONS, MNS and CNS dose analyses, which is a change to the AST for these plants. These gap fraction calculations used a projected power history that bounds the current ONS, MNS and CNS core design limits which is in accordance with RG 1.183, Table 3, Footnote 11.

Because increased gap fractions were determined by Duke Energy and a change to the AST was made, dose consequences were reanalyzed for fuel handling-type accidents. Dose consequences were not reanalyzed for other non-fuel-handling accidents since no fuel rod that is predicted to enter departure from nucleate boiling (DNB) will be permitted to operate beyond the limits of RG 1.183, Table 3, Footnote 11. The revised dose consequences for ONS, MNS and CNS continue to satisfy the requirements set forth in 10 CFR 50.67 and the acceptance criteria set forth in RG 1.183, Section 4.4.

10 CFR 50.71(e)

Requirements for updating a facility's final safety analysis report (FSAR) are in 10 CFR 50.71, "Maintenance of Records, Making of Reports." The regulations in 10 CFR 50.71(e) require that the FSAR be updated to include all changes made in the facility or procedures described in the FSAR and all safety evaluations performed by the licensee in support of requests for license amendments. Per RG 1.183, the analyses required by 10 CFR 50.67 are subject to this 10 CFR 50.71(e) requirement. Therefore, the affected radiological analyses descriptions in the FSAR will be updated to reflect the proposed changes included with this amendment. The descriptions of superseded analyses will be removed from the FSAR in the interest of maintaining a clear design basis for ONS, MNS and CNS.

4.2 Precedent

The NRC has previously approved changes similar to the proposed changes in this License Amendment Request for other nuclear power plants including:

1. Prairie Island Nuclear Generating Plant: Application dated January 20, 2004 (ADAMS Accession No. ML040270067); NRC Safety Evaluation dated September 10, 2004 (ADAMS Accession No. ML042430504).

Similar to Prairie Island, Duke Energy chose to follow the RG 1.183, Table 3, Footnote 11 alternative to calculate gap release fractions using bounding power histories and NRC approved methodology. Also similar to Prairie Island, Duke Energy's analysis results show the radiological consequences of the fuel handling-type accidents remain within the regulatory dose acceptance criteria contained in RG 1.183, both for personnel offsite and operators in the control room.

2. Three Mile Island, Unit 1: Application dated January 23, 2001 (ADAMS Accession No. ML010300215); Supplement dated August 22, 2001 (ADAMS Accession No. ML012400035); NRC Safety Evaluation dated April 30, 2002 (ADAMS Accession No. ML021080289).

   Similar to AmerGen with Three Mile Island, Unit 1, Duke Energy postulates fuel assemblies in future fuel cycle designs at Oconee, McGuire and Catawba that have the potential to exceed the 6.3 kW/ft Linear Heat Generation Rate limit in RG 1.183, Table 3, Footnote 11 for burnups greater than 54 GWD/MTU.

4.3 <u>No Significant Hazards Consideration Determination</u>

Duke Energy is requesting an amendment to change the Updated Final Safety Analysis Reports (UFSAR) for Catawba Nuclear Station (CNS), Units 1 and 2; McGuire Nuclear Station (MNS), Units 1 and 2; and Oconee Nuclear Station (ONS), Units 1, 2 and 3. Specifically, Duke Energy is requesting the Nuclear Regulatory Commission's approval of gap release fractions for high-burnup fuel rods (i.e., greater than 54 GWD/MTU) that exceed the 6.3 kW/ft linear heat generation rate (LHGR) limit detailed in Table 3, Footnote 11 of Regulatory Guide (RG) 1.183. Duke Energy proposes an alternative set of non-Loss of Coolant Accident (LOCA) gap release fractions using a projected power history that bounds the current CNS, MNS and ONS reactor core design limits in order to support the request. Finally, the dose consequences contained in the CNS, MNS and ONS UFSARs for fuel handling-type accidents are proposed to be updated in order to reflect damaged fuel assemblies that contain fuel rods operating above the 6.3 kW/ft LHGR limit in RG 1.183.

Duke Energy has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of Amendment," as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed change involves using gap release fractions for high-burnup fuel rods (i.e., greater than 54 GWD/MTU) that exceed the 6.3 kW/ft linear heat generation rate (LHGR) limit detailed in Table 3, Footnote 11 of RG 1.183. Increased gap release fractions were determined and accounted for in the dose analysis for Catawba Nuclear Station (CNS), Units 1 and 2; McGuire Nuclear Station (MNS), Units 1 and 2; and Oconee Nuclear Station (ONS), Units 1, 2 and 3. The dose consequences reported in each site's Updated Final Safety Analysis Report (UFSAR) were reanalyzed for fuel handling-type accidents only. Dose consequences were not reanalyzed for other non-fuel-handling accidents since no fuel rod that is predicted to enter departure from nucleate boiling (DNB) will be permitted to operate beyond the limits of RG 1.183, Table 3, Footnote 11. The current NRC requirements, as described in 10 CFR 50.67, specifies

dose acceptance criteria in terms of Total Effective Dose Equivalent (TEDE). The revised dose consequence analyses for fuel handling-type events at CNS, MNS and ONS meet the applicable TEDE dose acceptance criteria (specified also in RG 1.183). A slight increase in dose consequences is exhibited. However, the increase is not significant and the new TEDE results are below regulatory acceptance criteria.

The changes proposed do not affect the precursors for fuel handling-type accidents analyzed in Chapter 15 of the CNS, MNS or ONS UFSARs. The probability remains unchanged since the accident analyses performed and discussed in the basis for the UFSAR changes, involve no change to a system, structure or component that affects initiating events for any UFSAR Chapter 15 accident evaluated.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   Response: No.

   The proposed change involves using gap release fractions for high-burnup fuel rods (i.e., greater than 54 GWD/MTU) that exceed the 6.3 kW/ft LHGR limit detailed in Table 3, Footnote 11 of RG 1.183. Increased gap release fractions were determined and accounted for in the dose analysis for CNS, MNS and ONS. The dose consequences reported in each site's UFSAR were reanalyzed for fuel handling-type accidents only. Dose consequences were not reanalyzed for other non-fuel-handling accidents since no fuel rod that is predicted to enter departure from nucleate boiling (DNB) will be permitted to operate beyond the limits of RG 1.183, Table 3, Footnote 11.

   The proposed change does not involve the addition or modification of any plant equipment. The proposed change has the potential to affect future core designs for CNS, MNS and ONS. However, the impact will not be beyond the standard function capabilities of the equipment. The proposed change involves using gap release fractions that would allow high-burnup fuel rods (i.e., greater than 54 GWD/MTU) to exceed the 6.3 kW/ft LHGR limit detailed in Table 3, Footnote 11 of RG 1.183. Accounting for these new gap release fractions in the dose analysis for CNS, MNS and ONS does not create the possibility of a new accident.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does the proposed change involve a significant reduction in a margin of safety?**

   Response: No.

   The proposed change involves using gap release fractions for high-burnup fuel rods (i.e., greater than 54 GWD/MTU) that exceed the 6.3 kW/ft LHGR limit detailed in Table 3, Footnote 11 of RG 1.183. Increased gap release fractions were determined and accounted for in the dose analysis for CNS, MNS and ONS. The dose consequences reported in each site's UFSAR were reanalyzed for fuel handling-type accidents only. Dose consequences were not reanalyzed for other non-fuel-handling accidents since no fuel rod that is predicted to enter departure from nucleate boiling (DNB) will be permitted to operate beyond the limits of RG 1.183, Table 3, Footnote 11.

   The proposed change has the potential for an increased postulated accident dose at CNS, MNS or ONS. However, the analysis demonstrates that the resultant doses are

within the appropriate acceptance criteria. The margin of safety, as defined by 10 CFR 50.67 and Regulatory Guide 1.183, has been maintained. Furthermore, the assumptions and input used in the gap release and dose consequences calculations are conservative. These conservative assumptions ensure that the radiation doses calculated pursuant to Regulatory Guide 1.183 and cited in this license amendment request are the upper bounds to radiological consequences of the fuel handling-type accidents analyzed. The analysis shows that with increased gap release fractions accounted for in the dose consequences calculations there is margin between the offsite radiation doses calculated and the dose limits of 10 CFR 50.67 and acceptance criteria of Regulatory Guide 1.183. The proposed change will not degrade the plant protective boundaries, will not cause a release of fission products to the public and will not degrade the performance of any structures, systems and components important to safety.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, Duke Energy concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of "no significant hazards consideration" is justified.

4.4 Conclusions

In conclusion, based on the considerations discussed above: (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.

5. ENVIRONMENTAL CONSIDERATION

The proposed amendment does not involve a significant hazards consideration, a significant change in the types of any effluents that may be released offsite, a significant increase in the amount of any effluents that may be released offsite or a significant increase in the individual or cumulative occupational radiation exposure. Although there is an increase in the amount of calculated radioactivity released, this increase is not considered significant because the new dose consequences of the fuel handling-type accidents analysis remain below the acceptance criteria specified in 10 CFR 50.67 and Regulatory Guide 1.183. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6. REFERENCES

1. Regulatory Guide 1.183, *Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors*, Revision 0, U.S. Nuclear Regulatory Commission, July 2000.
2. Draft Regulatory Guide DG-1199, *Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors (Proposed Revision 1 of Regulatory Guide 1.183)*, U.S. Nuclear Regulatory Commission, October 2009.

3. ANSI/ANS-5.4-1982, *Method for Calculating the Fractional Release of Volatile Fission Products from Oxide Fuel*, American National Standard published by the American Nuclear Society, November 1982.

4. ANSI/ANS-5.4-2011, *Method for Calculating the Fractional Release of Volatile Fission Products from Oxide Fuel*, American National Standard published by the American Nuclear Society, May 2011.

5. PNNL-18212, *Update of Gap Release Fractions for non-LOCA Events Utilizing the Revised ANS 5.4 Standard,* Revision 1, Pacific Northwest National Laboratory (C. Beyer and P. Clifford), June 2011.

6. NUREG/CR-2507, *Background and Derivation of ANS-5.4 Standard Fission Product Release Model,* Compiled by Southern Science Applications Inc. for the U.S. Nuclear Regulatory Commission, January 1982.

7. NUREG/CR-7003, *Background and Derivation of ANS-5.4 Standard Fission Product Release Model*, J. Turnbull and C. Beyer for the U.S. Nuclear Regulatory Commission, January 2010.

8. ORNL/TM-2005/39 (Version 5.1) *SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation*, Oak Ridge National Laboratory, November 2006.

9. BAW-10231P-A, Revision 1, *Copernic Fuel Rod Design Computer Code*, Framatome ANP (now Areva), January 2004.

10. WCAP-15063-P-A, Revision 1 with Errata, *Westinghouse Improved Performance Analysis and Design Model (PAD 4.0)*, July 2000.

11. NRC Information Notice IN 2009-23, *Nuclear Fuel Thermal Conductivity Degradation*, October 8, 2009.

12. NRC letter, *McGuire Nuclear Station, Units 1 and 2, Issuance of Amendments Regarding Implementation of Alternative Source Term Methodology (TAC Nos. MC9746 and MC9747)*, December 22, 2006 (ADAMS Accession No. ML063100406).

13. NRC letter, *McGuire Nuclear Station, Units 1 and 2, Issuance of Amendments Regarding Adoption of the Alternate Source Term Radiological Analysis Methodology (TAC Nos. MD8400 and MD8401)*, March 31, 2009 (ADAMS Accession No. 090890627).

14. NRC letter, *Catawba Nuclear Station, Units 1 and 2 RE: Issuance of Amendments (TAC Nos. MB3758 and MB3759)*, April 23, 2002 (ADAMS Accession No. ML021140431).

15. NRC letter, *Catawba Nuclear Station, Units 1 and 2 RE: Issuance of Amendments (TAC Nos. MB7014 and MB7015)*, September 30, 2005 (ADAMS Accession No. ML052730312).

16. NRC letter, *Oconee Nuclear Station, Units 1, 2 and 3 RE: Issuance of Amendments (TAC Nos. MB3537, MB 3538, and MB3539)*, June 1, 2004 (ADAMS Accession No. ML041540097).

17. LTR-NRC-12-27, *Westinghouse Input Supporting Licensee Response to NRC 10 CFR 50.54(f) Letter Regarding Nuclear Fuel Thermal Conductivity Degradation*, Westinghouse (J. Gresham) to U.S. Nuclear Regulatory Commission, March 7, 2012 (ADAMS Accession No. ML12072A035).

18. NRC letter, *Prairie Island Nuclear Generating Plant, Units 1 and 2 – Issuance of Amendments RE: Selective Implementation of Alternate Source Term for Fuel Handling*

Accidents (TAC Nos. MC1843 and MC1844), September 10, 2004 (ADAMS Accession No. ML042430504).

19. NRC letter, *Prairie Island Nuclear Generating Plant, Units 1 and 2 – Issuance of Amendments RE: Adoption of Alternative Source Term Methodology (TAC Nos. ME2609 and ME2610)*, January 22, 2013 (ADAMS Accession No. ML112521289).

Enclosure 2

Proposed Updated Final Safety Analysis Report Changes (Mark-up)

An asterisk (*) is used to identify markups that are part of previously approved UFSAR changes. They are not associated with the proposed changes described in this LAR.

Chemical Group	% Core Inventory
Am, Cm)	

The core inventory of fission products and actinides was calculated with the computer SCALE computer code system (Reference 24) and specifically, the SAS2H module.

The SAS2H module uses the BONAMI and NITAWL codes to perform resonance self-shielding corrections. (The NITAWL code uses the Nordheim Integral Treatment to process cross section data having resonance parameters.) SAS2H uses the XSDRNPM code to perform the transport calculations to develop cell-flux-weighted cross sections for use in the subsequent depletion calculations with ORIGEN-S. (The XSDRNPM code uses the method of discrete ordinates to perform neutron transport calculations through the pin-cell model.) The ORIGEN-S code is used by SAS2H to perform depletion calculations throughout the burnup period modeled by the input. The multigroup calculations performed with BONAMI, NITAWL, and XSDRNPM yield cross sections that are collapsed to a one-group library for use in ORIGIN-S (a point depletion code). In the calculation enveloping values are taken for burnup, enrichment, irradiation, and power level (102% rated power or 3479 MWth). The core isotopic inventories of fission products are presented in Table 15-12.

Inventory in the Fuel Pin Gap: For some design basis accidents, fuel damage as predicted in bounding analyses is limited to fuel clad failure. These include the design basis locked rotor accident, rod ejection accident, fuel handling accidents, and weir gate drop. For these design basis accidents, the radioactive source term is limited to the activity initially in the gap between the fuel pellets and fuel clad. The percentage of the core fission product inventory in the fuel pin gaps (gap fractions) for the non LOCA design basis accidents are as follows (Reference 26):

	Gap Fraction for non LOCA DBAs (%)	
Chemical Group	Rod Ejection	Other DBAs
I-131	10	~~10~~ 8 *
Kr-85	10	~~8~~ 10
Other noble gases (Kr, Xe)	10	5
Other halogens (Br, I)	10	5
Alkali metals (Rb, Cs)	12	12

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The values for core inventory in the gap conform to the germane regulatory position except in the cases of bromine and alkali metals following a rod ejection accident. No values for these fission products for the rod ejection accident are given in the regulatory position. The gap fractions for bromine are assumed to be the same as for iodine. The gap fractions for alkali metals are taken to be the same for the rod ejection and locked rotor accidents.

The isotopic inventories for a fuel assembly used in the AST analysis of the design basis locked rotor and rod ejection accidents was calculated with the SCALE computer code suite. These calculations account for radial peaking, setting the radial peaking to limiting values. It also sets the burnup, enrichment, irradiation, and power level to enveloping values. The fuel assembly fission product inventories used in the analyses of the design basis locked rotor and rod ejection accidents are presented in Table 15-73. The fuel assembly fission product inventories used in the analyses of the design basis fuel handling accident and weir gate drop are presented in Table 15-45.

Residual Decay Heat

Total Residual Heat: Residual heat in a subcritical core is calculated for the loss of coolant accident per the requirements of Appendix K of 10CFR 50. These requirements include assuming infinite irradiation

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For the analyses of non-DNB accidents and for those fuel pins which are operated so as to exceed the rod power/burnup criteria of Footnote 11 in RG 1.183, the gap fractions from RG 1.183 are increased by a factor of 3 for Kr-85, Xe-133, Cs-134 and Cs-137, and increased by a factor of 2 for I-131, and other noble gases, halogens and alkali metals (References 27 & 28). The fuel cycle design ensures that none of these fuel pins experience DNB following any design basis accident.

10. American Nuclear Society, "Decay Heat Power in Light Water Reactors", *ANSI/ANS 5.1*, 1979.

11. EPRI, "RETRAN-02: A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems", *EPRI NP-1850-CCM*, Revision 6, December 1995.

12. EPRI, "VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores", *EPRI NP-2511-CCM-A*, Revision 3, August 1989.

13. Deleted Per 1994 Update

14. Deleted Per 1994 Update

15. Deleted Per 1994 Update

16. "Nuclear Design Methodology using CASMO-4/SIMULATE-3 MOX", DPC-NE-1005-P-A, Revision 1, November 2008.

17. Calvo, R., "Setpoint Study: Duke Power Company Catawba Nuclear Plant Units 1 and 2", WCAP-10041-P, March, 1984.

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19. DiNunno, J. J., et al, "Calculation of Distance Factors for Power and Test Reactor Sites", TID-14844, March, 1962.

20. Bell, M. J., "ORIGEN - The ORNL Isotope Generation and Depletion Code", ORNL-4628, May, 1973.

21. Radiation Shielding Information Center, Oak Ridge National Laboratory, "ORIGEN Yields and Cross Sections - Nuclear Transmutation and Decay Data from ENDF/B-IV", RSIC-DLC-38, September, 1975.

22. Deleted Per 2006 Update

23. Deleted Per 2006 Update

24. Oak Ridge National Laboratory, SCALE: A Modular Code System for Performing Standardized Analyses for Licensing Evaluation, NUREG/CR-0200, USNRC, March 1997.

25. "Duke Power Company Westinghouse Fuel Transition Report," DPC-NE-2009P-A, Revision 2, December 18, 2002.

26. USNRC, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, Regulatory Guide 1.183.

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27. This is a placeholder for the "Table 3 Footnote 11" (T3F11) license amendment request (LAR). This LAR will be identified upon NRC approval of it.

28. This is a placeholder for the NRC letter of approval and enclosed Safety Evaluation of the T3F11 LAR. It will be identified upon its publication.

Section 15.0

of core life immediately proceeding shutdown. The gap model discussed in ~~Regulatory Guide 1.183~~ is used to determine the fuel-clad gap activities. Thus 10 percent of the total assembly iodines and noble gases, except for 30 percent of Kr-85, are assumed to be in the fuel-clad gap. The total assembly and fuel-clad gap activities are given in Table 15-45.

15.7.4.2.1 Postulated Fuel Handling Accident Outside Containment

The analyses of a postulated fuel handling accident are performed as follows:

1. A conservative analysis was completed with the method of Alternative Source Terms and in conformance to R.G. 1.183 (References 1 and 2)

Deleted per 2004 update.

The parameters used for each of these analyses are listed in Table 15-46.

The basis for the Regulatory Guide 1.183 evaluation is as follows:

1. The accident is assumed to occur 72 hrs after plant shutdown.
2. All of the rods in one fuel assembly are ruptured.
3. The damaged assembly is the highest powered assembly in the core region to be discharged. The values for individual fission product inventories in the damaged assembly are calculated assuming full power operation at the end of core life immediately preceding shutdown. The r ~~Radial peaking factors were obtained as follows: A generic calculation was performed to determine fuel temperatures for Westinghouse RFA fuel as a function of axial and radial position and burnup. This analysis made use of a pin power history that provides upper bounds for power for all fuel pins in the core. Following procedures, the power history in the generic analysis is confirmed to be the upper bound the power history for all pins in the core.~~ is set to 1.65.
4. The maximum fuel rod pressurization is ≤ 1300 psig.
5. The minimum water depth between the top of the damaged fuel rods and the spent fuel pool surface is 23 ft.
6. All of the gap activity in the damaged rods is released to the spent fuel pool and consists of 5 percent of the total noble gases other than krypton-85, 10 percent of the krypton-85, 5 percent of the total radioactive ~~iodine~~ halogens other than iodine-131, and 8 percent of iodine-131 in the rods at the time of the accident. Insert B ~~The source term assumed for the Fuel Handling Accidents and Weir Gate Drop include only iodine and noble gas radioisotopes. Regulatory Position 1.2 of Regulatory Guide 1.183 mandates that the gap activity source term for the fuel handling accidents (including the weir gate drop) include alkaline metals (cesium and rubidium radioisotopes) and bromine isotopes as well as noble gas and iodine isotopes. However, it is acceptable to assume that the alkali metals are retained in the spent fuel pool (or reactor cavity). This is equivalent to omitting alkali metals from the source term for these accidents. Bromine radioisotopes also are excluded from the gap activity assumed available for release following the fuel handling accident (and weir gate drop). The half life of the bromine radioisotopes are small compared to the lower value taken for the decay time prior to the initiating event (3 days). With one exception, this is also true of the daughter isotopes. That exception is Kr-85, the daughter of beta decay of Br-85. The half life of Kr-85 is 10.756 years. However, the activity of Kr-85 produced from decay of Br-85 (and Br-85m) is very small compared to the activity of Kr-85 assumed to be in the isotopic inventory for the affected fuel assembly (Table 15-45). For the above reasons, it is acceptable to limit the radiological source terms for the fuel handling accidents (and weir gate drop) to noble gas and iodine radioisotopes.~~
7. ~~Noble gases released~~ to the spent fuel pool are then immediately released at ground level to the environment.

Insert B:

For those fuel pins which exceed the rod power/burnup criteria of Footnote 11 in RG 1.183, the gap fractions from RG 1.183 are increased by a factor of 3 for Kr-85, Xe-133, Cs-134 and Cs-137, and increased by a factor of 2 for I-131, and other noble gases, halogens and alkali metals.

8. The iodine gap inventory is composed of diatomic iodine (99.85 percent) and organic species (0.15 percent).
9. The spent fuel pool effective decontamination factor for iodine is 200. The composiion fractions for the iodine leaving the spent fuel pool are 57% for diatomic iodine and 43% for organic iodine compounds.
10. Noble gases are not held up in the fuel pool water.
11. The radioisotopes in the gap were assumed to be released instantly to the spent fuel pool. The timing of the release of radioactivity from the spent fuel pool to the environment was assumed to take the profile of an exponential decay function. It was assumed that 98% of all radioactivity released to the environment would be released within 2 hours.
12. Atmospheric dispersion conditions are assumed to be the 0-2 hour ground level case. The potential release paths for the Fuel Handling Accident outside containment are either through penetrations from the Fuel Building to the yard or the unit vent stack either via the Fuel Building Ventilation System (no credit taken for the VF filters) or via the Auxiliary Building (no credit taken for filtration or holdup or mixing). The release path was taken to be the unit vent stack as it is associted with the higher set of control room atmospheric dispersion factors (X/Q's).
13. Maximum burnup is 62 GWD/MTU. ~~For burnup less than 54 GWD/MTU, the linear heat rate is less than 6.3 kW/ft.~~
14. No credit is taken for filtration by the VF System. No credit is taken for mixing within the fuel building.
15. Only one Control Room Area Ventilation (VC) System outside air intake is taken to be open for the duration of the releases, the one exposed to the contaminated air. during *
16. Offsite power is assumed to be available ~~uring~~ the course of the accident.
17. The on-line VC filter train is assumed to fail at the initiating event. Since this is not a Safety Injection event and offsite power is assumed to be available, the standby VC filter train does not automatically start. Manual start of the standby VC filter train with a 30 minute delay is assumed.

15.7.4.2.2 Postulated Fuel Handling Accident Inside Containment

The possibility of a fuel handling accident inside Containment during refueling is relatively small due to the many physical, administrative, and safety restrictions imposed on refueling operations. Nevertheless, consideration is given to one accident; a drop of a fuel assembly into the refueling cavity by the refueling machine inside Containment. The impact would result in breaching of the fuel rod cladding and release of a portion of the fission gases from the damaged fuel rods to the refueling cavity.

The parameters used are listed in Table 15-46.

The analysis of the radiological consequences of the fuel handling accident inside containment also is conducted pursuant to R.G. 1.183. The basis for evaluation of the fuel handling accident inside containment is the same as that for the evaluation for the fuel handling accident outside containment (Items 1-14 listed in Section 15.7.4.2.2). In particular, the following clarification is made.

1. No credit is taken for filtration of releases by the Containment Purge Ventilation (VP) System. No credit is taken for containment isolation or mixing in containment. The process of release is assumed to same as that posed for the fuel handling accident outside containment: release of the gap activity to the reactor cavity, release from the reactor cavity, and timed release to the environment. The particulars of this process are exactly the same as assumed for the evaluation of the fuel handling

15.7.4.3.2 Postulated Fuel Handling Accident Inside Containment

Based on the foregoing model, the TEDE at the EAB is estimated to be 1.1 Rem. This is significantly below the acceptance criterion for this accident of 6.3 Rem listed in R.G. 1.183. The TEDE to the control room operators is conservatively estimated to be 1.6 Rem. This is well below the acceptance criterion of 5 Rem listed in R.G. 1.183.

Replace with Insert A

15.7.4.3.3 Postulated Weir Gate Drop

Based on the foregoing model, the total effective dose equivalent at the EAB is estimated to be 1.7 Rem. The acceptance criterion of 6.3 cited in R.G. 1.183 for fuel handling accidents is applied to the evaluation of the weir gate drop. The limiting TEDE at the EAB is significantly below this acceptance criterion. The TEDE to the control room operators is conservatively estimated to be 1.6 Rem. This is significantly below the acceptance criterion of 5 Rem listed in R.G. 1.183.

Replace with Insert B

15.7.5 Spent Fuel Cask Drop Accident

Analyses have been completed to determine the consequences of three scenarios in which a fuel cask is either tipped or dropped at Catawba.

In the first scenario, a fuel cask is either tipped or dropped toward a spent fuel pool. The analysis of this scenario is reported in Section 9.1.2.3. It was concluded that a dropped or tipped cask could not fall into the spent fuel pool. Accordingly, this scenario does not yield any radiological consequences.

The second scenario involves dropping a loaded fuel cask onto the ground at Catawba. In this scenario, the loaded cask is sealed. The analysis demonstrates that the fuel cask will not breach on impact. Accordingly, this scenario also yelds no radiological consequences.

The third scenario involves a drop of a loaded but unsealed fuel cask into the fuel cask pit. Since this cask was not sealed, it was assumed to be breached on impact. It was assumed that the dropped cask contained 37 fuel assemblies that have been cooled for 4 years. It also was assumed that the fuel assemblies are submerged at all times during this scenario. The analysis of this fuel cask drop scenario was completed with the method of Alternative Source Terms (AST) and in conformance to Regulatory Guide 1.183 and Appendix B. The fuel assembly isotopic activities (assuming no prior decay) include ^{85}Kr (29,320 Ci) and its two precursors: ^{85}Br (517,800 Ci) and ^{85m}Kr (519,100 Ci). Of all isotopes in the resulting source term, only ^{85}Kr escapes the water in the fuel cask pit; the other radioisotopes either disappear with radioactive decay or remain in the pool (^{134}Cs and ^{137}Cs). Rupture of the pin cladding for all fuel assemblies in the dropped cask and release of gap activity into the water in the pit was assumed. The entire ^{85}Kr inventory was assumed to be released to the environment within 2 hours of event initiation. The analysis included a sensitivity study to ensure the calculation of the limiting radiation dose in the control room. Insert C

The acceptance criteria for radiation doses for the fuel cask drop included 6.3 Rem at the offsite locations and 5.0 Rem in the control room. The acceptance limit for the offsite radiation doses is in accord with Standard Review Plan 15.7.5, NUREG-0612 Section 5.1, 10 CFR 50.67, and Regulatory Guide 1.183. The limit for control room radiation doses is taken pursuant to 10 CFR 50.67 and General Design Criterion 19. The limiting radiation doses from the analysis of this fuel cask drop are listed in Table 15-14.

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Insert C:

For those fuel pins which exceed the rod power/burnup criteria of Footnote 11 in RG 1.183, the gap fractions from RG 1.183 are increased by a factor of 3 for Kr-85, Xe-133, Cs-134 and Cs-137, and increased by a factor of 2 for I-131, and other noble gases, halogens and alkali metals.

Insert C1:

15.7.6 References

1. This is a placeholder for the "Table 3 Footnote 11" (T3F11) license amendment request (LAR). This LAR will be identified upon NRC approval of it.

2. This is a placeholder for the NRC letter of approval and enclosed Safety Evaluation of the T3F11 LAR. It will be identified upon its publication.

Table 15-14. Total Effective Dose Equivalents (TEDEs - Rem) Following Design Basis Events

Design Basis Event	UFSAR Section	Exclusion Area Boundary	Low Population Zone Boundary	Control Room
Main Steam Line Break	15.1.5.3			
Pre-existent iodine spike		0.13	0.03	
Accident initiated iodine spike		0.14	0.13	
Locked Rotor Accident	15.3.3.3	1.63	0.31	1.56
Rod Ejection Accident	15.4.8.3	4.75	3.37	2.70
Instrument Line Break	15.6.2			
Pre-existent iodine spike		0.60	0.09	0.80
Accident initiated iodine spike		0.17	0.02	0.18
Steam Generator Tube Rupture (1)	15.6.3.3			
Pre-existent iodine spike		1.19	0.28	1.32
Accident initiated iodine spike		0.61	0.19	0.81
Loss of Coolant Accident	15.6.5.3	8.79	3.78	3.31
Waste Gas Tank Rupture	15.7.1	0.27	0.04	0.10
Liquid Storage Tank Rupture	15.7.2	0.99	0.14	1.00
Fuel Handling Accident (3,4)	15.7.4	~~1.0~~ 1.76		~~2.3~~ 2.59
Weir Gate Drop (4)	15.7.4	~~2.9~~ 2.68		~~3.6~~ 4.24
Fuel Cask Drop (4)	15.7.5	~~0.005~~ 0.006		0.001

Notes:

1) A supplemental analysis of the steam generator tube rupture is reported in Section 15.6.3.3. It is the basis for a set of conditions cited in Facility Operating License Amendment 159/151. The license conditions set limits of 0.46 µCi/gm Dose Equivalent Iodine-131 (DEI) for equilibrium iodine specific activity in the reactor coolant and 26 DEI for transient iodine specific activity in the reactor coolant. Thyroid radiation doses are calculated in this supplemental analysis and reported in Table 15-74.

Table 15-45. Activities in Highest Inventory Discharged Assembly for Postulated Fuel Handling Accidents

Isotope	Assembly Activity (curies)	Fraction of Activity in Gap[1]	Gap Activity (curies)
I-130	2.32E04	0.05	1.16E03
I-131	7.46E05	0.08	5.97E04
I-132	1.10E06	0.05	5.50E04
I-133	1.61E06	0.05	8.05E04
I-134	1.86E06	0.05	9.30E04
I-135	1.53E06	0.05	7.65E04
I-136	7.91E05	0.05	3.96E04
Xe-131m	3.97E03	0.05	1.99E02
Xe-133m	4.71E04	0.05	2.36E03
Xe-133	1.53E06	0.05	7.65E04
Xe-135m	2.84E05	0.05	1.42E04
Xe-135	3.36E05	0.05	1.68E04
Xe-137	1.46E06	0.05	7.30E04
Xe-138	1.51E06	0.05	7.55E04
Kr-83m	1.25E05	0.05	6.25E03
Kr-85m	2.80E05	0.05	1.40E04
Kr-85	6.78E03	0.10	6.78E02
Kr-87	5.76E05	0.05	2.88E04
Kr-88	8.14E05	0.05	4.07E04
Kr-89	1.05E06	0.05	5.25E04

Note:

1. NRC Assumption in Regulatory Guide 1.183

2. Insert E as Note 2

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Insert D

Nuclide	Assembly Inventory (curies)	Gap Release Fractions[1]	Gap Inventory (curies)
Br-83	1.31E+05	0.05	6.55E+03
Br-85	2.99E+05	0.05	1.50E+04
Br-87	4.95E+05	0.05	2.48E+04
I-130	3.95E+04	0.05	1.98E+03
I-131	8.09E+05	0.08	6.47E+04
I-132	1.18E+06	0.05	5.90E+04
I-133	1.67E+06	0.05	8.35E+04
I-134	1.95E+06	0.05	9.75E+04
I-135	1.60E+06	0.05	8.00E+04
Kr-83m	1.32+05	0.05	6.60E+03
Kr-85m	2.98E+05	0.05	1.49E+04
Kr-85	7.48E+03	0.10	7.48E+02
Kr-87	6.15E+05	0.05	3.08E+04
Kr-88	8.69E+05	0.05	4.35E+04
Kr-89	1.12E+06	0.05	5.60E+04
Xe-131m	1.24E+04	0.05	6.20E+02
Xe-133m	5.20E+04	0.05	2.60E+03
Xe-133	1.65E+06	0.05	8.25E+04
Xe-135m	3.62E+05	0.05	1.81E+04
Xe-135	4.12E+05	0.05	2.06E+04
Xe-137	1.55E+06	0.05	7.75E+04
Xe-138	1.59E+06	0.05	7.95E+04
Rb-86	2.54E+03	0.12	3.05E+02
Rb-88	8.89E+05	0.12	1.07E+05
Rb-89	1.18E+06	0.12	1.42E+05
Rb-90	1.12E+06	0.12	1.34E+05
Cs-134	2.06E+05	0.12	2.47E+04
Cs-136	5.92E+04	0.12	7.10E+03
Cs-137	9.23E+04	0.12	1.11E+04
Cs-138	1.66E+06	0.12	1.99E+05
Cs-139	1.58E+06	0.12	1.90E+05

Insert E:

For those fuel pins which exceed the rod power/burnup criteria of Footnote 11 in RG 1.183, the gap fractions from RG 1.183 are increased by a factor of 3 for Kr-85, Xe-133, Cs-134 and Cs-137, and increased by a factor of 2 for I-131, and other noble gases, halogens and alkali metals.

6.	Data for conversion of activity to dose		
	a.	Coefficients for committed effective dose equivalents	FGR-11
	b.	Coefficients for deep dose equivalent	FGR-12
	c.	Other data for conversion of activity to dose	R.G. 1.183
7.	Total Effective Dose Equivalent (TEDE's – Rem)		Note 6
	a.	EAB TEDE	~~1.6~~ 1.76
	b.	Control room TEDE	~~3.3~~ 2.59

Notes:

1. The data and assumptions listed in Table 15-46 are applied to the fuel handling accident inside containment and the fuel handling accident outside containment.
2. ~~Alkali (cesium and rubidium) and bromine isotopes are not included in the source term. Cf. Section 15.7.4.1 for the justification.~~ Replace with Insert F
3. The term "pool" denotes the spent fuel pool for fuel handling accidents outside containment and the reactor cavity for fuel handling accidents inside containment.
4. This time constant was selected to ensure that at 98% of the iodine taken to be released from the pool would be released to the environment within 2 hours.
5. Only the values for the first time period (0-8 hr for the LPZ X/Q and 0-2 hr for the control room X/Q are listed as the release of radioactivity to the environment is essentially complete in 2 hours. Cf. Note 4.
6. The LPZ TEDE is not listed as it is less than the EAB TEDE and compared to the same acceptance criterion (6.3 Rem).

Insert F:

For those fuel pins which exceed the rod power/burnup criteria of Footnote 11 in RG 1.183, the gap fractions from RG 1.183 are increased by a factor of 3 for Kr-85, Xe-133, Cs-134 and Cs-137, and increased by a factor of 2 for I-131, and other noble gases, halogens and alkali metals.

Table 15-58. Parameters for Postulated Weir Gate Drop

1. Data and Assumption	
a. Decay time before the initiating event (days)	19.5
b. Number of fuel assemblies impacted	7
c. All other data and assumptions	Table 15-46
2. Total Effective Dose Equivalent	Note 1
a. EAB TEDE	~~2.9~~ 2.68
b. Control room TEDE	~~3.6~~ 4.24

Note:

1) The LPZ TEDE is not listed as it is less than the EAB TEDE and compared to the same acceptance criterion (6.3 Rem).

15.7.3 Postulated Radioactive Releases Due to Liquid Tank Failures

Based on the discussion in Section 2.4.12 and from an evaluation of radioactive releases from a postulated liquid tank failure it is concluded that the effluent from a failed liquid tank at McGuire could not directly enter groundwater. Furthermore, the radioactive concentrations at the nearest potable water intake from a liquid tank failure that follows a surface water pathway are determined to be small fractions of the 10 CFR Part 20, Appendix B limits for effluent concentrations.

15.7.4 Fuel Handling Accidents in the Containment and Spent Fuel Storage Buildings

15.7.4.1 Fuel Handling Accident Inside Containment

15.7.4.1.1 Identification of Causes and Accident Description

A fuel handling accident is postulated whereby a fuel assembly is damaged while being moved inside containment. Activity is released from the fuel assembly into the water in the fuel transfer canal and then into the containment atmosphere. The activity released into containment is released to the environment through the assumed open equipment hatch. Control room operators manually start control room outside air filtered pressurization within a prescribed period.

A fuel handling accident inside containment is classified as an ANS Condition IV event, a limiting fault. See Section 15.0 for a discussion of Condition IV events.

15.7.4.1.2 Analysis of Effects and Consequences

INSERT A

Method of Analysis

Fission product inventories are listed in Table 15-35 and were determined as described in Section 15.0.

The isotopics of the damaged assembly bound all spent fuel and design types. The modeled assembly isotopics are based upon bounding fuel characteristics (burnup, enrichment, peaking, etc.). It is assumed to have decayed at least three days prior to being damaged. The gap inventory from all of the fuel pins is assumed to be instantaneously released into the water. As the radioactivity moves up through the water to the surface, particulates (radioiodines and alkali metals) are removed and elemental iodine is scrubbed. The remainder of the radioiodines and all of the noble gases enter the containment atmosphere and are released through the equipment hatch in containment when it is open. This is the most limiting release point and it bounds other potential release locations. Neither containment purge filtration nor any other filtration is credited once the activity is released from the fuel transfer canal water to the environment. The control room model credits filtration by the Control Room Ventilation System and includes unfiltered in-leakage. Although the majority of the activity is released over a two hour period, the modeled release is continued for a 30 day period for control room and low population zone doses.

The fuel handling accident in containment and the fuel handling accident inside the spent fuel pool building are the same postulated accidents, but with different atmospheric dispersion factors.

The offsite release and dose consequences for this accident are calculated using methods described in Regulatory Guide 1.183.

Results

The calculated offsite doses are listed in Table 15-12.

15.7.4.1.3 Environmental Consequences

The doses from this accident are within 10 CFR 50.67 limits.

Insert A:

For fuel pins which exceed the rod power/burnup criteria of Footnote 11 in RG 1.183, the gap fractions from RG 1.183 are increased by a factor of 3 for Kr-85, Xe-133, Cs-134 and Cs-137, and increased by a factor of 2 for I-131, and other noble gases, halogens and alkali metals (References 1 and 2).

15.7.4.5 Spent Fuel Cask Drop in Fuel Building

15.7.4.5.1 Identification of Causes and Accident Description

The spent fuel cask drop accident scenario postulates that a dry storage cask is dropped when moving the fuel from the spent fuel pool to the dry storage area. Activity is released from the fuel pins to the canister. From there, the activity is released from the cask to the atmosphere. The activity released into the Fuel Building is released to the environment via the Fuel Handling Ventilation System.

15.7.4.5.2 Analysis of Effects and Consequences

Method of Analysis

The fuel assembly isotopics for this accident assume seven years decay time. Table 15-60 contains a list of these isotopics. Thirty-two fuel assemblies inside the cask are ruptured when the dry cask is dropped in transit from the spent fuel pool to the dry storage area. Activity is assumed to be released from the fuel into the cask cavity as provided in Interim Staff Guidance - 5 (ISG-5). Much of the activity released into the cask is deposited in the cask while some is released to the fuel building. The fuel building ventilation system collects the airborne activity released into the building, filters it through HEPA and charcoal filter banks, and releases it to the environment.

More detailed assumptions are found in Table 15-59.

15.7.4.5.3 Environmental Consequences

The doses from this accident are within a small fraction of the 10 CFR 100 limits.

The calculated offsite doses are listed in Table 15-12.

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15.7.5 References

1. This is a placeholder for the "Table 3 Footnote 11" (T3F11) license amendment request (LAR). This LAR will be identified upon NRC approval of it.

2. This is a placeholder for the NRC letter of approval and enclosed Safety Analysis of the T3F11 LAR. It will be identified upon its publication.

15.7.4.2 Fuel Handling Accident in the Fuel Building - Fuel Assembly Drop

15.7.4.2.1 Identification of Causes and Accident Description

A fuel handling accident in the Fuel Building is postulated whereby a fuel assembly is damaged while being moved inside the Spent Fuel Pool Building. Activity is released from the fuel assembly into the water in the spent fuel pool and then into the Fuel Building atmosphere. The activity released into the Fuel Building is released to the environment through the unit vent (via the Fuel Handling Ventilation System). Control room operators manually start control room outside air filtered pressurization within a prescribed period.

A fuel handling accident inside the fuel building is classified as an ANS Condition IV event, a limiting fault. See Section 15.0 for a discussion of Condition IV events.

15.7.4.2.2 Analysis of Effects and Consequences

Method of Analysis

INSERT B

The isotopics of the damaged assembly bound all spent fuel and design types. The modeled assembly isotopics are based upon bounding fuel characteristics (burnup, enrichment, peaking, etc.). It is assumed to have decayed at least three days prior to being damaged. The gap inventory from all of the fuel pins is assumed to be instantaneously released into the water. As the radioactivity moves up through the water to the surface, particulates (radioiodines and alkali metals) are removed and elemental iodine is scrubbed. The remainder of the radioiodines and all of the noble gases enter the Fuel Building atmosphere and are released through the unit vent. This is the most limiting release point and it bounds other potential release locations. Neither Fuel Handling Ventilation filtration nor any other filtration is credited once the activity is released from the spent fuel pool water to the environment. The control room model credits filtration by the Control Room Ventilation System and includes unfiltered in-leakage. Although the majority of the activity is released over a two hour period, the modeled release is continued for a 30 day period for control room and low population zone doses.

The fuel handling accident in the spent fuel pool building and the fuel handling accident inside containment are the same postulated accidents, but with different atmospheric dispersion factors.

Results

~~The calculated offsite doses are listed in Table 15-12.~~ INSERT C

15.7.4.2.3 Environmental Consequences

The doses from this accident are within 10 CFR 50.67 limits.

15.7.4.3 Fuel Handling Accident in the Fuel Building - Weir Gate Drop

15.7.4.3.1 Identification of Causes and Accident Description

The weir gate drop accident scenario postulates that one of the two weir gates in the spent fuel pool is dropped, while being maneuvered, onto seven fuel assemblies. Activity is released from the fuel assemblies into the water in the spent fuel pool and then into the Fuel Building atmosphere. The activity released into the Fuel Building is released to the environment through the unit vent (via the Fuel Handling Ventilation System). Control room operators manually start control room outside air filtered pressurization within a prescribed period.

Insert B:

For fuel pins which exceed the rod power/burnup criteria of Footnote 11 in RG 1.183, the gap fractions from RG 1.183 are increased by a factor of 3 for Kr-85, Xe-133, Cs-134 and Cs-137, and increased by a factor of 2 for I-131, and other noble gases, halogens and alkali metals.

Insert C:

The results of the fuel handling accident inside the spent fuel building is bounded by the fuel handling accident inside containment.

15.7.4.3.2 Analysis of Effects and Consequences

Method of Analysis

INSERT D

The bounding fuel assembly isotopics used in the single fuel assembly Fuel Handling Accidents is used for each of the seven fuel assemblies damaged in the weir gate drop. These assemblies are decayed the minimum amount of time after refueling which must pass until the weir gate is permitted to move (at least 17.5 days). The gap inventory from all of the fuel pins in all seven of the assemblies is assumed to be instantaneously released into the spent fuel pool water. As the radioactivity moves up through the water to the surface, particulates (radioiodines and alkali metals) are removed and elemental iodine is scrubbed. The remainder of the radioiodines and all of the noble gases enter the Fuel Building atmosphere and are released through the unit vent. This is the most limiting release point and it bounds other potential release locations. Neither Fuel Handling Ventilation filtration nor any other filtration is credited once the activity is released from the spent fuel pool water to the environment. The control room model credits filtration by the Control Room Ventilation System and includes unfiltered in-leakage. Although the majority of the activity is released over a two hour period, the modeled release is continued for a 30 day period for control room and low population zone doses.

The offsite release and dose consequences for this accident were calculated using methods described in Regulatory Guide 1.183.

15.7.4.3.3 Environmental Consequences

The doses from this accident are within 10 CFR 50.67 limits.

The calculated offsite doses are listed in Table 15-12.

15.7.4.4 Spent Fuel Cask Drop into the Spend Fuel Pool

Based on an analysis discussed in Section 9.1.2.3.2 it is concluded that the cask can not enter the spent fuel pool due to a postulated dropping or tipping of the cask. Based on this conclusion, the radiological consequences of a spent fuel cask drop accident need not be evaluated.

Radiation doses were calculated for a drop of a loaded but unsealed fuel cask into the fuel cask pit. Since this cask was not sealed, it was assumed to be breached on impact. It was assumed that the dropped cask contained 37 fuel assemblies that have been cooled for 4 years. It also was assumed that the fuel assemblies are submerged at all times during this scenario. The analysis of this fuel cask drop scenario was completed with the method of Alternative Source Terms (AST) and in conformance to Regulatory Guide 1.183 and Appendix B. The fuel assembly isotopic activities (assuming no prior decay) include ^{85}Kr (29,320 Ci) and its two precursors: ^{85}Br (517,800 Ci) and ^{85m}Kr (519,100 Ci). Of all isotopes in the resulting source term, only ^{85}Kr escapes the water in the fuel cask pit; the other radioisotopes either disappear with radioactive decay or remain in the pool (^{134}Cs and ^{137}Cs). Rupture of the pin cladding for all fuel assemblies in the dropped cask and release of gap activity into the water in the pit was assumed. INSERT E The entire ^{85}Kr inventory was assumed to be released to the environment within 2 hours of event initiation. The analysis included a sensitivity study to ensure the calculation of the limiting radiation dose in the control room.

The acceptance criteria for radiation doses for the fuel cask drop included 6.3 Rem at the offsite locations and 5.0 Rem in the control room. The acceptance limit for the offsite radiation doses is in accord with Standard Review Plan 15.7.5, NUREG-0612 Section 5.1, 10 CFR 50.67, and Regulatory Guide 1.183. The limit for control room radiation doses is taken pursuant to 10 CFR 50.67 and General Design Criterion 19. The limiting radiation doses from the analysis of this fuel cask drop are listed in Table 15-12.

Insert D:

For fuel pins which exceed the rod power/burnup criteria of Footnote 11 in RG 1.183, the gap fractions from RG 1.183 are increased by a factor of 3 for Kr-85, Xe-133, Cs-134 and Cs-137, and increased by a factor of 2 for I-131, and other noble gases, halogens and alkali metals.

Insert E:

For fuel pins which exceed the rod power/burnup criteria of Footnote 11 in RG 1.183, the gap fractions from RG 1.183 are increased by a factor of 3 for Kr-85, Xe-133, Cs-134 and Cs-137, and increased by a factor of 2 for I-131, and other noble gases, halogens and alkali metals.

15.10 Formerly Chapter 15 Appendix B - Supplementary Radiological Analyses

15.10.1 Tornado Missile Impact - Spent Fuel Analysis

15.10.1.1 Identification of Causes and Accident Description

The north end of the fuel building at McGuire is not provided with tornado missile protection. The radiological consequences of a tornado missile impact are mitigated by limiting the age of the fuel discharged to the spent fuel pool.

15.10.2 Analysis of Effects and Consequences

INSERT F

15.10.3 Method of Analysis

An analysis was performed to determine the dose consequences in the event of a postulated tornado missile accident. A tornado-generated missile is assumed to breach the fuel building and rupture twelve assemblies in region 1 of the spent fuel pool. The entire gap activity is assumed to be instantly released to the spent fuel pool water. The activity is assumed to be released to the envirnoment, with no holdup in the fuel building, over a 2-hour period. Key assumptions pertaining to this accident include:

1. The accident is assumed to occur 5 days after reactor shutdown.
2. Fuel pins are breached and are equivalent to twelve assemblies in region 1 being ruptured.
3. The source term is listed in Table 15-35.
4. The effective iodine decontamination factor is 200. Only elemental iodine and organic iodine compounds are released from the spent fuel pool. All particulates are assumed to stay in the water.
5. The Fuel Building Ventilation (VF) System is credited for this analysis in the sense that the release point is not the unit vent, but is actually the rollup door, since it is not tornado-proof. No credit is taken however for filtration from the VF system.
6. The Control Room Area Ventilation (VC) System is assumed to be in service within 30 minutes of the accident. The rate of unfiltered inleakage is set to 500 cfm before pressurization and 210 cfm once pressurization occurs.
7. Other assumptions are detailed in Table 15-39.

Results

A maximum number of fuel pins that rupture, equivalent to 12 fuel assemblies, could potentially suffer a loss of integrity due to a tornado missile accident. The resulting doses are presented in Table 15-12.

15.10.3.1 Environmental Consequences

The doses for this accident are below 10 CFR 50.67 limits.

Insert F:

For fuel pins which exceed the rod power/burnup criteria of Footnote 11 in RG 1.183, the gap fractions from RG 1.183 are increased by a factor of 3 for Kr-85, Xe-133, Cs-134 and Cs-137, and increased by a factor of 2 for I-131, and other noble gases, halogens and alkali metals.

Accident	FSAR Section	Rem Total Effective Dose Equivalent (TEDE)		
		Exclusion Area Boundary	Low Population Zone	Control Room
Loss of Coolant Accident	15.6.5	12.25	2.23	4.86
		(25.0)	(25.0)	(5.0)

Accident	FSAR Section	Rem Total Effective Dose Equivalent (TEDE)		
		Exclusion Area Boundary	Low Population Zone	Control Room
Waste Gas Decay Tank Failure	15.7.1	0.25	0.02	0.01
		(0.5)	(0.5)	(5.0)
Liquid Storage Tank Failure	15.7.2	1.89	0.17	0.61
		(2.5)	(2.5)	(5.0)

Accident	FSAR Section	Rem Total Effective Dose Equivalent (TEDE)		
		Exclusion Area Boundary	Low Population Zone	Control Room
Cask Drop in Pit	15.7.4	~~8.97E-03~~ 0.01	~~7.97E-04~~ 0.0009	~~4.68E-04~~ 0.0006
		(6.3)	(6.3)	(5.0)

Accident	FSAR Section	Rem Total Effective Dose Equivalent (TEDE) Exclusion Area Boundary	Low Population Zone	Control Room
Fuel Handling Accident				
Inside Containment	15.7.4.1	~~2.95~~ 3.25	~~0.27~~ 0.29	~~3.52~~ 3.86
		(6.3)	(6.3)	(5.0)
~~Inside SFP Building~~	~~15.7.4.3~~	~~2.95~~	~~0.27~~	~~1.46~~
		~~(6.3)~~	~~(6.3)~~	~~(5.0)~~
Dropped Weir Gate Inside SFP Building	15.7.4.3	~~5.60~~ 6.16	~~0.51~~ 0.56	~~2.97~~ 3.25
		(6.3)	(6.3)	(5.0)

Accident	FSAR Section	2-hr Dose at 2500 ft. Exclusion Area Boundary	30 day Dose at 29000 ft. Low Population Zone	
		Exclusion Area Boundary	Low Population Zone	Control Room
Tornado Generated Missile Accident	15.10.3	~~2.58~~ 2.84	~~2.59~~ 2.86	~~4.25~~ 4.66
		(25.0)	(25.0)	(5.0)

Accident	FSAR Section	2-hr Dose at 2500 ft. Exclusion Area Boundary	
		Whole Body	Thyroid
Cask Drop Accident	15.7.4.5	0.01	0.2
		(2.5)	(30.0)

Nuclide	Assembly Inventory (curies)	Gap Release Fractions[1]	Gap Inventory (curies)
Cs-139	1.58E+06	0.12	1.90E+05

Note:

NRC Assumption in Regulatory Guide 1.183

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Insert G:

For fuel pins which exceed the rod power/burnup criteria of Footnote 11 in RG 1.183, the gap fractions from RG 1.183 are increased by a factor of 3 for Kr-85, Xe-133, Cs-134 and Cs-137, and increased by a factor of 2 for I-131, and other noble gases, halogens and alkali metals.

9. Electrical bus voltage and frequency control are credited. These are controlled by non-safety components.

10. The Integrated Control System trips both main feedwater pumps on a high steam generator level indication. A high level indication may occur following a main steam line break due to the pressure drops that result from the blowdown of the steam generator. Tripping of the main feedwater pumps will be assumed to occur in the steam line break analysis only if the plant response is more limiting.

15.1.10 Environmental Consequences Calculation Methodology

Environmental Consequences

A summary of the offsite doses is presented in Table 15-16. A description of each accident analysis is given in the appropriate section.

Fission Product Inventories

Inventory in the Core: Fission product inventories within the core are calculated based on the ORIGEN methodology (e.g., ORIGEN-ARP or SAS2H/ORIGEN-S of the SCALE computer code)(Section 15.1, Ref. 27). The core inventories for the Maximum Hypothetical Accident are shown in Table 15-15.

Insert A

Inventory in the Fuel Pellet Clad Gap: The fuel pin gap activities were determined using Regulatory Guide 1.183 (Section 15.1, Ref. 35). ~~For O2C25, the gap fractions from Regulatory Guide 1.183 were increased by a factor of 2 for all rods in all assemblies which contained rods that exceeded the rod power/burnup criteria in RG 1.183.~~ The environmental consequences of the control rod ejection accident, and fuel handing accidents are based on the assumption that the fission products in the gap between the fuel pellets and the cladding of the damaged fuel rods are released as a result of cladding failure. The inventories used for the control rod cluster assembly ejection accident are shown in Table 15-50. The gap inventory for the fuel handling accident is shown in Table 15-1.

Inventory in the Reactor Coolant: The quantity of fission products released to the reactor coolant during steady state operation is based on the use of escape rate coefficients (sec^{-1}) derived from experiments involving purposely defected fuel elements. (Section 15.1, References 29, 30, 31, 32) These coefficients represent the fraction of the activity in the fuel that is released, per unit time. Values of the escape rate coefficients used in the calculations are shown in Table 11-4.

Calculations of isotopic specific activities in the reactor coolant arising from steady-state fission product releases from the fuel (except for Kr-85) were performed with the Duke computer code PWR-SOURCE. The code calculates equilibrium reactor coolant fission product inventories and specific activities from the steady-state solutions to the differential equations for the radioactive decay chains for more than 150 isotopes. Due to the extremely long half life of Kr-85, an equilibrium activity level will not be reached in the reactor coolant during an operating cycle. For this particular isotope, the activity level is calculated from the exact solution of the decay chain, utilizing equilibrium activities of parent isotopes as inputs.

The reactor coolant activity levels are listed in Table 15-51. Dose Equivalent Iodine (DEI) and Dose Equivalent Xenon (DEX) calculations are shown in Table 15-65 and Table 15-66.

Inventory in the OTSGs and Secondary-Side Systems: The concentration of the iodine isotopes in the steam generators and secondary system coolant are assumed to be at the Technical Specification limit of 0.1 μCi/gm dose equivalent I-131, unless otherwise stated in a specific accident analysis. No credit is taken for removal of iodine from the secondary coolant by station demineralizers.

The concentrations of noble gases in the secondary side coolant are assumed to be negligible, and therefore are not modeled. Noble gases entering the secondary coolant system are continuously vented to the atmosphere via the condenser off-gas system. Thus, there would be only very small quantities of

Insert A:

For non-DNB fuel pins which exceed the rod power/burnup criteria of Footnote 11 in RG 1.183, the gap fractions from RG 1.183 are increased by a factor of 3 for Kr-85, Xe-133, Cs-134 and Cs-137, and increased by a factor of 2 for I-131, and other noble gases, halogens and alkali metals (References 46 and 47). The fuel cycle design ensures that none of these fuel pins experience DNB following any design basis accident.

25. American National Standard for Decay Heat Power in Light Water Reactors, ANSI/ANS-5.1-1979, American Nuclear Society, August 1979.

26. Deleted per 2008 Update.

27. NUREG/CR-0200, "SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation."

28. Deleted per 2009 Update.

29. Frank, P. W., et al., Radiochemistry of Third PWR Fuel material Test - X-1 Loop NRX Reactor, WAPD-TM-29, February, 1957.

30. Eichenberg, J. D., et al, Effects of Irradiation on Bulk UO2, WAPD-183, October, 1957.

31. Allison, G. M., and Robertson, R. F. S., The Behavior of Fission Products in Pressurized-Water Systems. A Review of Defect Tests on UO2 Fuel Elements at Chalk River, AECL-1338, 1961.

32. Allison, G. M., and Roe, H. K., The Release of Fission Gases & Iodines from Defected UO2 Fuel Elements of Different Lengths, AECL-2206, June, 1965.

33. Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10CFR 50, Appendix I," Rev. 1, October 1977.

34. The Code of Federal Regulations, Title 10, Part 100, Section 11 (10CFR 100.11), "Determination of Exclusion Area, Low Population Zone, and Population Center Distance."

35. Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 20, 2000.

36. Deleted per 2009 Update.

37. Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel, BAW-10227-PA, Revision 1, June 2003.

38. RETRAN-3D - A program for transient Thermal-Hydraulic Anlaysis of Complex Fluid Flow Systems, Electric Power Research Institute, November 2009.

39. The Code of Federal Regulations, Title 10, Part 50, Section 67 (10CFR 50.67), "Accident Source Term."

40. DPC-NE-2015-PA, Oconee Nuclear Station Mark-B-HTP Fuel Transition Methodology, Revision 0, Duke Power, October 2008

41. BHTP DNB Correlation Applied with LYNXT, BAW-10241(P)(A), Revision 1, Framatome ANP, July 2005

42. A Correlation of Rod Bundle Critical Heat Flux for Water in the Pressure Range 150 to 725 psia, IN-1412, Idaho Nuclear Corporation, July 1970.

43. Regulatory Guide 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," June 2003.

44. CASMO-4 Fuel Assembly Burnup Program User's Manual, SSP-01/400 Rev. 5, Studsvik Scandpower, June 2007

45. Oconee Nuclear Design Methodology using CASMO-4/SIMULATE-3, DPC-NE-1006-PA, SER dated August 2, 2011.

Insert A1

Insert A1:

46. This is a placeholder for the "Table 3 Footnote 11" (T3F11) license amendment request (LAR). This LAR will be identified upon NRC approval of it.

47. This is a placeholder for the NRC letter of approval and enclosed Safety Evaluation of the T3F11 LAR. It will be identified upon its publication.

15.11 Fuel Handling Accidents

15.11.1 Identification of Accident

Spent fuel assemblies are handled entirely under water. The Core Operating Limits Report, refueling boron concentration, ensures shutdown margin is maintained. Procedures ensure that fuel assemblies are in configurations such that this shutdown margin is maintained In the spent fuel storage pool, the fuel assemblies are stored under water in storage racks with a minimum boron concentration as specified by the Core Operating Limits Report (COLR) in the pool water. Under these conditions, a criticality accident during refueling is not considered credible. Fuel handling consists of all fuel assembly shuffling and transfer operations between the reactor, the spent fuel pool, the fuel shipping casks, and dry storage transfer cask. Mechanical damage to the fuel assemblies during transfer operations is possible but improbable. The mechanical damage type of accident is considered the maximum potential source of activity release during refueling operations.

15.11.2 Analysis and Results

15.11.2.1 Base Case Fuel Handling Accident in Spent Fuel Pool

During fuel handling operations, it is possible that a fuel assembly can be dropped, causing mechanical damage with a subsequent release of fission products. To conservatively evaluate the offsite dose consequences of such an accident, conservative assumptions are made. The following analysis assumes the accident occurs within the spent fuel pool building.

Insert B

The fuel assembly gap inventory is assumed to contain a fission product inventory from a maximum burned fuel assembly at a radial peaking factor of 1.65. The gap fractions used are from Reg. Guide 1.183 and the reactor has been shutdown for 72 hours, which is the minimum time for RCS cooldown, reactor closure head removal, and removal of the first fuel assembly. ~~For O2C25, the gap fractions from Regulatory Guide 1.183 were increased by a factor of 2 for all rods in the assembly which contained rods that exceeded the rod power/burnup criteria in RG 1.183.~~ The actual isotopic curie contents are listed in Table 15-1. It is also assumed that all 208 fuel pins are mechanically damaged such that the entire gap inventory is released to the surrounding water. Since the fuel pellets are cold, only the gap inventory is released. The maximum fuel rod internal pressure in the spent fuel is 1300 psig as used in the computer code TACO3 to determine the fuel rod internal pressure.

The gases released from the damaged fuel assembly pass upward through the spent fuel pool water prior to reaching the Auxiliary Building atmosphere. Noble gases are assumed to not be retained in the pool water. According to Reg Guide 1.183, an iodine decontamination factor of 200 can be used for water depths of 23 feet or greater. Since the spent fuel pool racks are at an elevation of 816.5 feet and the minimum water level in the Spent Fuel Pool is equal to or greater than 837.84 feet, there is a minimum of 21.34 feet of water over the fuel storage racks, including instrument error. An experimental test program (Reference 2) evaluated the extent of removal of iodine released from a damaged irradiated fuel assembly. Iodine removal from the released gas takes place as the gas rises through the water. The extent of iodine removal is determined by mass transfer from the gas phase to the surrounding liquid and is controlled by the bubble diameter and contact time of the bubble with the water. The following analytical expression is given as a result of this experimental test program:

Iodine Decontamination Factor (DF) = $73\ e^{0.313\,(t/d)}$

Where:

t = bubble rise time, seconds

Insert B:

For fuel pins which exceed the rod power/burnup criteria of Footnote 11 in RG 1.183, the gap fractions from RG 1.183 are increased by a factor of 3 for Kr-85, Xe-133, Cs-134 and Cs-137, and increased by a factor of 2 for I-131, and other noble gases, halogens and alkali metals.

d = effective bubble diameter, cm

Since the minimum water depth over a dropped fuel assembly is less than 23 feet (21.34 feet), the assumed iodine DF must be less than 200, according to Reg. Guide 1.183, and calculated with comparable conservatism. Using the above relationship, with a water depth of 21.34 feet, a comparable DF is equal to 183 (Revision 1).

Deleted paragraph(s) per 2006 update.

The activity released from the water's surface is released within a two-hour period as a ground release. The atmospheric dilution is calculated using the two-hour ground release dispersion factor of 2.2×10^{4} sec/m^3.

The total effective dose equivalent (TEDE) doses are given in Table 15-16. These values are below the limits given in Regulatory Guide 1.183.

15.11.2.2 Base Case Fuel Handling Accident Inside Containment

The offsite dose consequences for a fuel handling accident inside containment were evaulated per the guidance given in Reg. Guide 1.183. Since the shallow end of the fuel transfer canal is at an elevation of 816.5 feet, the same iodine decontamination factor used for the Fuel Handling Accident in the Spent Fuel Pool is used for the Fuel Handling Accident inside Containment. The activity released from the refueling water is released as a ground release, which has an atmospheric dispersion factor of 2.2×10^{4} sec/m^3. There is no credit taken for any containment closure/integrity resulting in the released activity from the refueling water going straight outside.

Using the fuel assembly gap inventory in Table 15-1, and assuming all 208 fuel pins are damaged, the calculated doses are appropriately within the guidelines given in Regulatory Guide 1.183. Insert C ~~For O2C25, the gap fractions from Regulatory Guide 1.183 were increased by a factor of 2 for all rods in the assembly which contained rods that exceeded the rod power/burnup criteria in RG 1.183.~~ The limiting doses for a fuel handling accident for a single fuel assembly event are given in Table 15-16.

15.11.2.3 Deleted Per 2006 Update

15.11.2.4 Shipping Cask Drop Accidents

Fuel shipping casks are used to transport irradiated fuel assemblies from the site and also between the Oconee 1 and 2 spent fuel pool and the Oconee 3 spent fuel pool.

Deleted paragraph(s) per 2006 update.

The worst case fuel handling accident sequence in which the fuel shipping cask impacts on the irradiated fuel assemblies in a spent fuel pool is evaluated. At no time is the cask suspended above the spent fuel; however, it is credible that with failure of the cask hoist cable that the cask, yoke, hook, and load block could, as a result of an eccentric drop, deflect and fall into the spent fuel pool and impact on top of the assemblies in the pool. The analysis is performed separately for the shared Unit 1 and 2 spent fuel pool and the Unit 3 spent fuel pool. In the first part of the analysis, the number of fuel assemblies damaged as a result of the cask drop is found. Subsequently the radiological consequences of the damaged assemblies are determined.

The following conservative assumptions are employed for determining the number of fuel assemblies damaged.

1. The cask, lifting yoke and load block are free to fall from elevation 844 ft., the top of the spent fuel pool, to elevation 816 ft. 5 in., the top of the fuel storage racks.

Insert C:

For fuel pins which exceed the rod power/burnup criteria of Footnote 11 in RG 1.183, the gap fractions from RG 1.183 are increased by a factor of 3 for Kr-85, Xe-133, Cs-134 and Cs-137, and increased by a factor of 2 for I-131, and other noble gases, halogens and alkali metals.

2. The drag on the cask, lifting yoke and load block from falling through 25.5 ft. of water is neglected.
3. The ability of the fuel storage cells to absorb energy beyond the point of elastic buckling is neglected.
4. The energy which is expended in deformation of the rack interconnecting members is neglected.
5. A deformed fuel storage cell results in the total loss of integrity of one fuel assembly.
6. The projected areas of the cask, lifting yoke and load block are oriented to contact the maximum number of fuel assemblies.

Using the above assumptions, the falling cask, lifting yoke, and load block will have 2.093 x 10^6 ft-lbf of kinetic energy at the instant of impact with the storage racks. This energy must be absorbed by the strain energy in the storage racks. For additional conservatism it is assumed that the storage racks which are directly impacted by the falling load in turn buckle and deflect into adjacent racks until the total energy of the falling cask is absorbed. The Unit 1 and 2 spent fuel pool contains 154 fuel storage positions under the direct impact area, with a total of 576 spent fuel assemblies which can potentially suffer a loss of integrity during a cask drop accident. The Unit 3 pool contains 156 fuel storage positions under the projected impact area, with a total of 518 assemblies which can be damaged during the accident. These analyses are based on the TN8 three element shipping cask.

Once the number of fuel assemblies which could be damaged is determined, dose analyses are performed which are consistent with Regulatory Guide 1.183, and NUREG0612. The following assumptions apply:

1. Spent fuel stored in the first 36 rows of the Unit 1 and 2 spent fuel pool closest to the spent fuel cask handling area has decayed at least 55 days. This is consistent with Technical Specification 3.7.15.a, "Plant Systems".
2. All fuel assemblies assumed damaged in excess of two full cores (354 assemblies) in the Unit 1 and 2 spent fuel pool are assumed to have decayed at least one year.
3. Spent fuel stored in the first 33 rows of the Unit 3 spent fuel pool closest to the spent fuel cask handling area has decayed at least 70 days. This is consistent with Technical Specification 3.7.15.b., "Plant Systems".
4. All fuel assemblies assumed damaged in excess of one full core (177 assemblies) in the Unit 3 spent fuel pool are assumed to have decayed at least one year.
5. The affected assemblies have the maximum core activity corresponding to a radial peaking factor of 1.2.
6. All rods of the affected assemblies are ruptured.
7. The iodine decontamination factor in pool water is 183.
8. There is no removal of activity by the spent fuel pool ventilation system filters prior to release to the environment.
9. Activity is released at ground level with an assumed χ/Q factor of 2.2 x 10^4 sec/m^3.
10. The fractions of noble gases and iodine in the gaps are shown below. ~~The gap fractions from Regulatory Guide 1.183 were increased by a factor of 2 for all rods in four assemblies for each cask drop scenario for O2C25 which contained four assemblies with rods that exceeded the rod power/burnup criteria in RG 1.183~~ (Reference 1).

Insert D

Kr85, I131	10%, 8%
All other noble gases	5%

Insert D:

For fuel pins which exceed the rod power/burnup criteria of Footnote 11 in RG 1.183, the gap fractions from RG 1.183 are increased by a factor of 3 for Kr-85, Xe-133, Cs-134 and Cs-137, and increased by a factor of 2 for I-131, and other noble gases, halogens and alkali metals.

Table 15-1. Reg. Guide 1.183 Fuel Handling Accident Source Term

Isotope	Gap Fraction (See Note 1)	72 Hour Gap Inventory (Ci/Fuel Assembly)
Kr-83m	0.05	2.34E-05
Kr-85m	0.10	3.72E-01
Kr-85	0.10	8.53E+02
Kr-87	0.05	2.39E-13
Kr-88	0.05	8.26E-04
Xe-131m	0.05	4.81E+02
Xe-133m	0.05	1.21E+03
Xe-133	0.05	5.26E+04
Xe-135m	0.05	5.38E+00
Xe-135	0.05	7.21E+02
Deleted Row(s) per 2009 Update		
I-131	0.08	4.35E+04
I-132	0.05	2.59E+04
I-133	0.05	6.44E+03
Deleted Row(s) per 2009 Update		
I-135	0.05	3.29E+01

Note:

1. ~~**For O2C25, the gap fractions from Regulatory Guide 1.183 were increased by a factor of 2 for all rods in all assembles which contained rods that exceeded the rod power/burnup criteria in RG 1.183.**~~

Insert E

Insert E:

For fuel pins which exceed the rod power/burnup criteria of Footnote 11 in RG 1.183, the gap fractions from RG 1.183 are increased by a factor of 3 for Kr-85, Xe-133, Cs-134 and Cs-137, and increased by a factor of 2 for I-131, and other noble gases, halogens and alkali metals.

Table 15-16. Summary of Transient and Accident Doses Including the Effects of High Burnup Reload Cores with Replacement Steam Generators

	Doses (rem)	
Fuel Handling Accident for Single Fuel Assembly Event		
TEDE at EAB	~~1.18~~ 1.33	
TEDE at LPZ	~~0.13~~ 0.14	
TEDE in Control Room	~~2.19~~ 2.45	
O2C25 Cycle-Specific Fuel Handling Accident for Single Fuel Assembly Event		
TEDE at EAB	2.35	
TEDE at LPZ	0.25	
TEDE in Control Room	4.38	
Steam Generator Tube Rupture	Case 1	Case 2
Thyroid at EAB	4.24E+1	2.80E+2
Whole body at EAB	1.46E-1	1.82E-1
Thyroid at LPZ	1.00E+1	6.93E+1
Whole body at LPZ	3.04E-2	4.00E-2
Waste Gas Tank Failure		
TEDE at EAB	4.4E-1	
Rod Ejection	Containment Release	Secondary Side Release
TEDE at EAB	5.23	2.66
TEDE at LPZ	2.00	1.35
TEDE in Control Room	4.46	4.92
Deleted row(s) per 2009 update		
Large Main Steam Line Break	Preaccident Iodine Spike	Concurrent Iodine Spike
TEDE at EAB	0.18	0.70
Deleted row(s) per 2012 Update		
TEDE at LPZ	0.05	0.22
TEDE in Control Room	0.76	1.30

	Doses (rem)	
Small Main Steam Line Break	Preaccident Iodine Spike	Concurrent Iodine Spike
TEDE at EAB	0.29	0.68
Deleted row(s) per 2012 Update		
TEDE at LPZ	0.06	0.24
TEDE in Control Room	1.29	1.69
Deleted row(s) per 2004 update		
Maximum Hypothetical Accident		
TEDE at EAB	10.86	
TEDE at LPZ	2.74	
TEDE in Control Room	4.39	
Deleted row(s) per 2008 update		
Fuel Cask Handling Accident for Multiple Fuel Assembly Event		
TEDE at EAB	~~1.85~~ 2.05	
TEDE at LPZ	~~0.20~~ 0.22	
TEDE in Control Room	~~3.65~~ 4.05	
Deleted row(s) per 2008 update		
Deleted row(s) per 2004 update		
Deleted row(s) per 2008 update		