DUKE ENERGY®

J. Ed Burchfield, Jr.
Vice President
Oconee Nuclear Station

Duke Energy
ON01VP | 7800 Rochester Hwy
Seneca, SC 29672

o: 864.873.3478
f: 864.873.5791
Ed.Burchfield@duke-energy.com

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

Duke Energy Carolinas, LLC
Oconee Nuclear Station (ONS), Units 1, 2, and 3
Docket Numbers 50-269, 50-270, and 50-287
Renewed Facility Operating License Nos. DPR-38, DPR-47, and DPR-55

Subject: License Amendment Request Proposing a Revised 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; License Amendment Request No. 2018-05

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. Catawba Nuclear Station, Units 1 and 2; McGuire Nuclear Station, Units 1 and 2; and Oconee Nuclear Station, Units 1, 2, and 3 -- Issuance of Amendments Regarding Request to Use an Alternate Fission Gas Gap Release Fraction, U.S. Nuclear Regulatory Commission, July 19, 2016.
3. 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.

Pursuant to 10 CFR 50.90, Duke Energy Carolinas, LLC (Duke Energy), hereby submits a license amendment request for Oconee Nuclear Station (ONS). This request for amendment would revise the facility 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). 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 ONS, and submitted herein, support an increase to the Reference 1 LHGR limit.

This ONS license amendment request is an update to a previous submittal that was approved in 2016 (Reference 2). This new amendment request applies the Reference 3 method exclusively,

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and would allow, for ONS, a higher bounding rod power history as well as the removal of the Reference 2 restriction on the number of rods per assembly (25) that can exceed the rod power/burnup criteria of Footnote 11 in Regulatory Guide 1.183.

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

The Enclosure to this letter provides an evaluation of the proposed changes. Applicable marked-up Updated Final Safety Analysis Report (UFSAR) pages are included in an Attachment. 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 the Enclosure.

Approval of the proposed license amendment is requested within one year of the date of this submittal. The amendment shall be implemented within 120 days following approval.

There are no new regulatory commitments contained in this letter.

In accordance with 10 CFR 50.91, a copy of this application is being provided to the State of South Carolina.

Inquiries on this proposed amendment request should be directed to Stephen C. Newman, Regulatory Affairs Lead Nuclear Engineer, at (864) 873-4388.

I declare under penalty of perjury that the foregoing is true and correct. Executed on November 1, 2018.

Sincerely,

J Ed Burchfield

J. Ed Burchfield, Jr.
Vice President
Oconee Nuclear Station

Enclosure: Evaluation of Proposed Change
Attachment: UFSAR Mark-Up
cc w/enclosure and attachment:

Ms. Catherine Haney, Administrator, Region II
U.S. Nuclear Regulatory Commission
Marquis One Tower
245 Peachtree Center Ave., NE, Suite 1200
Atlanta, GA 30303-1257

Ms. Audrey Klett, Project Manager
(by electronic mail only: audrey.klett@nrc.gov)
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Mail Stop O-8G9A
11555 Rockville Pike
Rockville, Maryland 20852

Mr. Eddy Crowe
NRC Senior Resident Inspector
Oconee Nuclear Station

Ms. Lynne Garner, Manager,
(by electronic mail only: garnerld@dhec.sc.gov)
Infectious and Radioactive Waste Management,
Bureau of Land and Waste Management
Department of Health & Environmental Control
2600 Bull Street
Columbia, SC 29201

ENCLOSURE

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

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1.0 SUMMARY DESCRIPTION

This technical evaluation supports a request to amend the Operating Licenses DPR-38, DPR-47, and DPR-55 for Oconee Nuclear Station (ONS), Units 1, 2, and 3.

The proposed changes would revise the dose consequences for the facility, as described in the ONS 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).

2.0 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:

> "[11]The release fractions listed here have been determined to be acceptable for use with currently approved LWR fuel with a peak burnup up to 62,000 MWD/MTU provided that the maximum linear heat generation rate does not exceed 6.3 kw/ft peak rod average power for burnups exceeding 54 GWD/MTU. As an alternative <u>[to the non-LOCA gap fractions in Table 3 and the limits of Footnote 11]</u>, 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. For the BWR rod drop accident and the PWR rod ejection accident, the gap fractions are assumed to be 10% for iodines and noble gases."

Note: Underlined text above is not part of RG 1.183 Footnote 11 but was added for clarity.

This follows a similar amendment request implemented in 2016 (Reference 10 [LAR submittal] and Reference 11 [NRC approval]). This LAR applies the method in the ANS 5.4 [2011] standard (Reference 4) exclusively, and would allow, for ONS, a higher bounding rod power history as well as the removal of the Reference 11 restriction on the number of rods per assembly (25) that can exceed the rod power/burnup criteria of Footnote 11 in Regulatory Guide 1.183.

Based on the evaluation provided in Section 3.1, Duke Energy proposes to revise the Reference 11 non-LOCA gap fractions for all high-burnup fuel rods (greater than 54 GWD/MTU) in each fuel assembly that operates in the ONS reactors. The changes are as follows:

- The values in Regulatory Guide 1.183, Table 3 will be increased by a factor of 4 (for ^{85}Kr, ^{134}Cs, and ^{137}Cs) which is consistent with the proposed UFSAR change.
- The values in Regulatory Guide 1.183, Table 3 are applied, without any multipliers, for all other radioisotopes.

These revised gap fractions allow LHGRs above 6.3 kW/ft for rod burnup greater than 54 GWD/MTU, as long as the LHGRs remain within the bounding power history evaluated in

Section 3.1. These higher LHGRs, up to the proposed limits, could be applied to all fuel rods in each fuel assembly operated in the ONS reactors.

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 Reference 4 method. In addition to the previous Oconee, McGuire, and Catawba submittal described earlier in this section, Duke Energy has also received approval for LARs related to increasing high-burnup LHGRs for H.B. Robinson Steam Electric Plant (Reference 12 [LAR] and Reference 13 [approval]) and Shearon Harris Nuclear Power Plant (Reference 14 [LAR] and Reference 15 [approval]).

As input to the gap fraction calculations, the approved fuel performance code COPERNIC (Reference 7) was employed to determine nodal fuel temperatures for rod burnups from 0 to 62 GWD/MTU. The COPERNIC temperature model accounts for thermal conductivity degradation effects. The fuel rods modeled with COPERNIC are associated with the 15x15 assembly type currently operating in the ONS reactors.

Section 3.2 includes an evaluation of the dose consequences of an ONS fuel handling-type accident. This includes the Fuel Handling Accident (single assembly event) and the Fuel Cask Handling Accident (multiple assembly event), in which the damaged fuel assemblies are composed of high-burnup fuel pins operated above 6.3 kW/ft. No other non-LOCA accidents that may result in departure from nucleate boiling (DNB) are considered (e.g., locked rotor accident, rod ejection accident, etc.). Fuel cycles for ONS are 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 changes proposed in this LAR would be reflected in updates to Sections 15.1.10, 15.11.2.1, 15.11.2.2, 15.11.2.4, and Tables 15-1 and 15-16 of the ONS UFSAR, which address design basis fuel handling accidents. Draft markups to the ONS UFSAR are provided in Enclosure 2.

3.0 TECHNICAL EVALUATION

Gap release fractions for high-burnup rods (greater than 54 GWD/MTU) with a revised allowable LHGR 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 ONS. The dose analysis is 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 the fuel handling accidents.

ONS has implemented the Alternative Source Term (AST) method in its current licensing basis (Reference 9), in accordance with Regulatory Guide 1.183. 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.

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. Nuclear Regulatory Commission (NRC) Information Notice 2009-23 (Reference 8) discusses this issue in more detail. Higher fuel temperatures will yield larger fission gas release fractions in the ANS 5.4 [1982] and [2011] models (References 3 and 4), particularly in the high-burnup range.

The ANS 5.4 [1982] standard has been revised, and the update (ANS 5.4 [2011]) acknowledges the conservatism of the previous version, based on additional experimental data after 1982. 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 employs the ANS 5.4 [2011] method exclusively, using an approved fuel performance code (COPERNIC, Reference 7). The gap release analysis accounts for TCD, and considers all pertinent long-lived and short-lived isotopes.

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

The limiting plant-specific power history, along with other parameters such as core average burnup, are provided in the reload design safety analysis report document that is prepared with each cyclical core design for ONS, and incorporated into the reload design, and this change does not impact the isotopic inventory for any accident besides the fuel handling type accidents (i.e., Fuel Handling and Fuel Cask Handling accidents).

3.1.1 Method

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 this standard are provided in References 5 and 6.

The method in the ANS 5.4 [2011] standard is 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 Fuel Rod Type Considered

The Framatome MkB-HTP 15x15 fuel rod design is considered for the fission gas release calculations. This is the design that is currently being irradiated in the ONS reactors. As this fuel type is representative of a general 15x15 pressurized water reactor (PWR) design, the analysis of the fuel rod is judged to be applicable to other 15x15 designs that may be used in the ONS reactors.

3.1.1.2 Rod Operational Power Histories

The core design must maintain fuel rod power peaking below the peaking analyzed in the dose analyses. Table 1 shows the rod powers that are used in the gap release analysis for ONS MkB-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 increment of 2.5 GWD/MTU), consistent with the restrictions of the ANS 5.4 [2011] method.

The ONS core average power (deposited within the fuel rod) is calculated below. The 0.973 value represents the fraction of total heat from fission that is deposited within the fuel rod.

ONS:

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

With this core average rod power, peaking factors can be determined from the rod powers in Table 1. The computational results of the gap fraction calculations are presented in Section 3.1.3.

Table 1. Projected Rod Powers in the Gap Release Analysis

ONS HTP fuel	
Rod Burnup Range (GWD/MTU)	Average Rod Power (kW/ft)
0 – 1	9.585
1 – 3	9.585
3 – 5	9.585
5 – 6	9.585
6 – 8	9.585
8 – 10	9.585
10 – 11	9.585
11 – 13	9.585
13 – 15	9.585
15 – 17	9.585
17 – 19	9.585
19 – 21	9.585
21 – 23	9.585
23 – 25.5	9.585
25.5 – 26	9.585
26 – 28	9.585
28 – 30	9.585
30 – 32	9.585
32 – 34	9.585

ONS HTP fuel	
Rod Burnup Range (GWD/MTU)	Average Rod Power (kW/ft)
34 – 36	9.585
36 – 38	9.585
38 – 40	9.585
40 – 42	9.585
42 – 44	9.585
44 – 46	9.585
46 – 48	9.585
48 – 50	9.585
50 – 52	9.385*
52 – 54	8.985*
54 – 55	8.685*
55 – 56	8.485*
56 – 57	8.285*
57 – 58	8.085*
58 – 59	7.885*
59 – 60	7.685*
60 – 61	7.485*
61 – 62	7.285*

* Values shown are at the midpoint of the pertinent burnup interval. The bounding LHGR from 50 to 62 GWD/MTU is a linear function: LHGR = 19.585 - 0.200*Burnup

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 the fuel handling accidents. Table 2 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 2. 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 [2011]

Gap fractions for each of the isotopes in Table 2 are determined using either a direct result from the 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 3.1.1.4.1 through 3.1.1.4.3 discuss the specific procedures. Short-lived isotope calculations require input nodal fuel temperatures and burnups for each time step listed in Table 1. These nodal inputs are produced by COPERNIC.

3.1.1.4.1 Long-Lived Nuclides ($T_{1/2}$ > 1 year)

The long-lived isotopes listed in Table 2 (Kr-85, Cs-134, and Cs-137) are treated as stable. The Kr-85 fission gas gap fraction is taken directly from the fuel performance code (COPERNIC), 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].

3.1.1.4.2 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:

$$\left(\frac{R}{B}\right)_{i,m} = \left(\frac{S}{V}\right)_{i,m} \sqrt{\frac{\alpha_n D_{i,m}}{\lambda_n}} \qquad (1)$$

where:

$$\left(\frac{S}{V}\right)_{i,m} = 120\ \mathrm{cm}^{-1} \ \text{ if } T_{i,m} \leq T_{link} \qquad (2)$$

$$\left(\frac{S}{V}\right)_{i,m} = 650\ \mathrm{cm}^{-1} \ \text{ if } T_{i,m} > T_{link} \qquad (3)$$

$\left(\frac{S}{V}\right)_{i,m}$ is the surface area to volume ratio for radial node *i* in axial node *m*

$T_{i,m}$ 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:

$$T_{link} = \frac{9800}{ln(176 \times Bu_m)} + 273 \ \text{ if } Bu_m \leq 18.2\ \mathrm{GWD/MTU} \qquad (4)$$

$$T_{link} = 1434 - (12.85 \times Bu_m) + 273 \ \text{ if } Bu_m > 18.2\ \mathrm{GWD/MTU} \qquad (5)$$

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 3

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

$$D_{i,m} = 7.6 \times 10^{-7} e^{-35000/T_{i,m}} + 1.41 \times 10^{-18} \dot{F}_m^{0.5} e^{-13800/T_{i,m}} + 2 \times 10^{-30} \dot{F}_m \qquad (6)$$

$$\dot{F}_m = 4 \times 10^{10} LHGR_m / \left(Diam_o^2 - Diam_i^2\right) \qquad (7)$$

$Diam_o$ 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 (1), values for the precursor variable α_n are provided for specific isotopes in ANS 5.4 [2011]. Pertinent precursor coefficients are shown in Table 3. 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. 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 nodal volumetric weighting if necessary. Any burnup dependence on short half-life isotopic inventories is ignored, as noted in item 4 of Section 3.1.2.

Table 3. 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

3.1.1.4.3 Remaining Short-Lived Nuclides ($T_{1/2}$ > 6 hours and $T_{1/2}$ < 1 year)

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

$$\left(\frac{R}{B}\right)_{i,m} = F_n \left(\frac{S}{V}\right)_{i,m} \sqrt{\frac{\alpha_{kr-85m} D_{i,m}}{\lambda_{kr-85m}}} \tag{8}$$

where:

$$F_n = \left(\frac{\alpha_n \lambda_{kr-85m}}{\lambda_n \alpha_{kr-85m}}\right)^{0.25} \tag{9}$$

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 (8) be used with I-132, even though its half-life is less than 6 hours, to account for the large pre-cursor effect of Te-132, which has a much longer half-life (3.2 days).

The diffusion coefficient in equation (8) is multiplied by a factor of 2 for any cesiums.

3.1.1.5 Computer Codes

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

- **COPERNIC** – this is a NRC-approved fuel performance code (Reference 7).
- **gapfrac** – this is a Visual Basic for Applications (VBA) program that computes fission gas gap fractions in accordance with ANS 5.4 [2011], using the method described in Section 3.1.1.4.

3.1.2 Assumptions / Calculation Bases

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

1) Nominal (best-estimate) fuel rod design/operational input was used for the COPERNIC model.

2) The rod power history selected for this analysis (see Table 1) bounds the limiting ONS power history, 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 (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 are judged to have a negligible effect on calculated release fractions.

5) For the gap fraction calculations, all fuel rod evaluations were performed using a sufficient number of equally-spaced axial fuel segments and equal-volume radial rings in the fuel pellet. The ANS 5.4 [2011] standard requires at least 7 equal-volume radial nodes, and 10 or more axial nodes for the gap fraction computations.

6) Fuel assembly axial power data from Reference 10 (for a recent ONS core design) were used to determine appropriate axial power shapes for the COPERNIC fuel performance code.

7) 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.

8) The gap fraction evaluation was performed only for ONS UO_2 fuel with no integral gadolinia poisons. Previous analysis (Reference 10) has shown that fuel rods with gadolinia poison yield lower fission gas release than rods of the same U-235 enrichment that do not contain gadolinia.

9) In accordance with ANS 5.4 [2011], gap fractions calculated using equations (1) and (8) in Section 3.1.1.4 are multiplied by a factor of 5, to account for uncertainties in release predictions.

3.1.3 Fission Gas Release Analysis -- Calculations / Results

Section 3.1.1 described the method that is used to compute gap fractions for an ONS HTP fuel rod with the power history profile shown in Table 1. The computer programs used for the calculations are discussed in Section 3.1.1.5.

The first step in the analysis was to build input decks for the COPERNIC code, so that appropriate nodal fuel temperatures could be obtained for input to the ANS 5.4 [2011] gap release equations. Input information for COPERNIC 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
- Enrichment and axial blanket details
- Reactor core operational data

Axial power shapes, as a function of rod burnup, were obtained from the ONS analysis performed in Reference 10. Using these shapes and the other input information detailed above, a COPERNIC case was executed for an ONS HTP fuel rod with a 5.00 wt % U-235 central fuel enrichment and 2.50 wt % U-235 axial blankets.

Next, the gap fractions for the short-lived Table 2 isotopes were calculated, using COPERNIC-computed fuel temperatures and the Table 1 power history. To perform the 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 **gapfrac** code was also used for the supporting calculations in the Reference 10, 12, and 14 submittals.

The ANS 5.4 [2011] fission gas release results from the **gapfrac** computations are shown in Figure 1, for selected short-lived isotopes. Table 4 lists the gap fractions calculated for each isotope from Table 2. The directly-computed maximum COPERNIC fission gas release yields the gap release value for Kr-85. As noted in subsection 3.1.1.4.1, 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 $\sqrt{2}$.

Based on the above discussion, as well as the Table 4 results below, it is can be seen that increased gap fractions for the long-lived isotopes must be accounted for in dose analyses if it is desired to exceed a 6.3 kW/ft LHGR above 54 GWD/MTU. The results of this analysis show that with the chosen power history in Table 1, calculated gap fractions remain below 4 times the Regulatory Guide 1.183 Table 3 values for Kr-85, Cs-134, and Cs-137. For the short-lived isotopes, Table 4 and Figure 1 show that the maximum computed gap fractions remain well under the existing Regulatory Guide 1.183 Table 3 values.

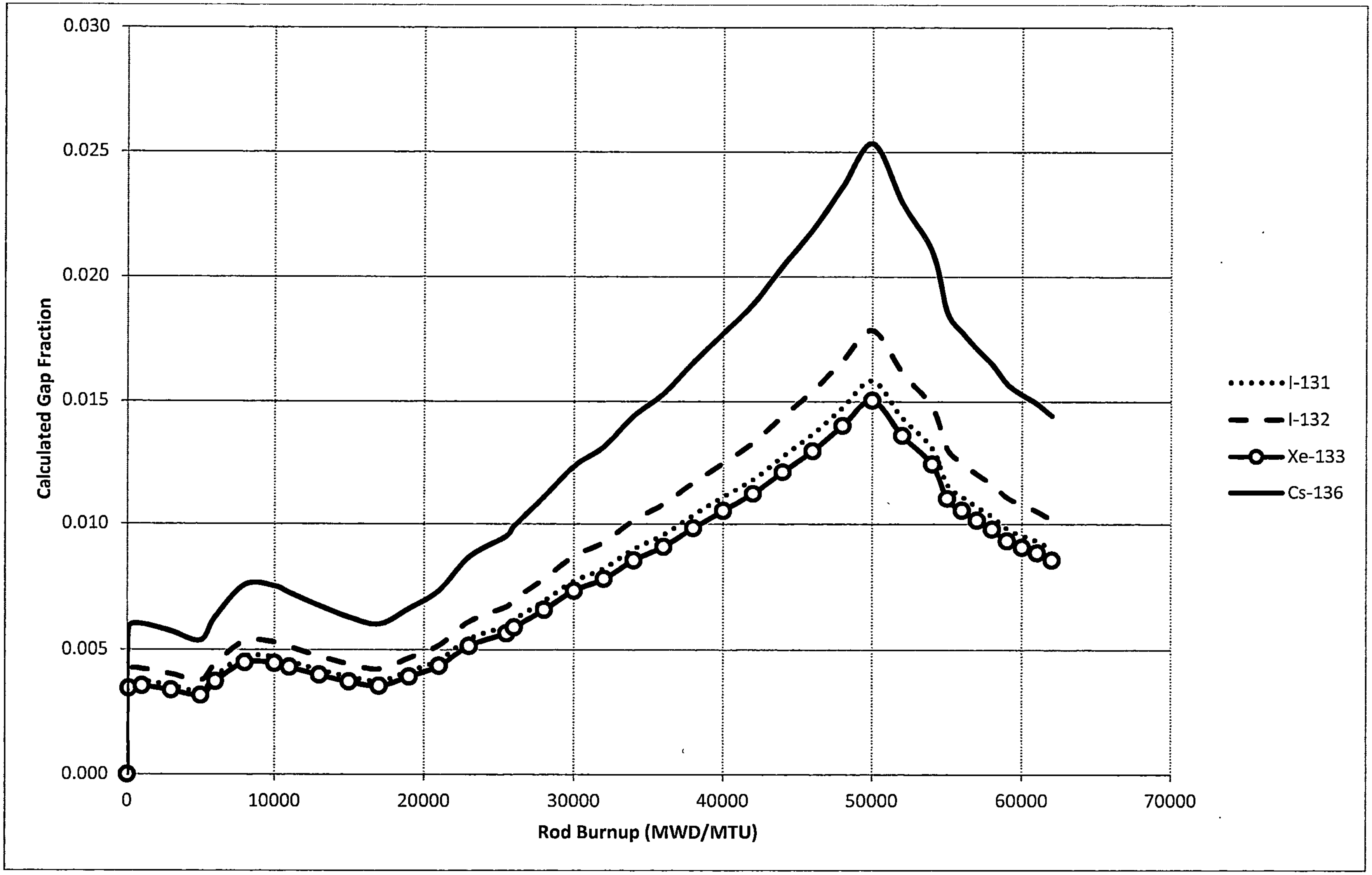

Figure 1. Calculated Gap Fractions for Selected Short-Lived Isotopes – 5.00 wt % U-235 ONS HTP Fuel

Table 4. Results from Gap Release Calculations

Isotope	Isotope Category	Reg Guide 1.183 Table 3 Value	ONS HTP fuel calculated maximum gap fraction	Ratio
Long-lived (> 1-yr half-life) Isotopes (from COPERNIC results)				
Kr-85	Kr-85	0.10	0.318	3.18
Cs-134	Alkali Metals	0.12	0.450	3.75
Cs-137	Alkali Metals	0.12	0.450	3.75
Short-lived (< 1-yr half-life) Isotopes (from gapfrac results)				
I-130	Other Halogens	0.05	0.0080	0.16
I-131	I-131	0.08	0.0158	0.20
I-132	Other Halogens	0.05	0.0179	0.36
I-133	Other Halogens	0.05	0.0095	0.19
I-134	Other Halogens	0.05	0.0054	0.11
I-135	Other Halogens	0.05	0.0068	0.14
Br-83	Other Halogens	0.05	0.0042	0.08
Br-85	Other Halogens	0.05	0.0006	0.01
Br-87	Other Halogens	0.05	0.0003	0.01
Kr-83m	Other Nobles	0.05	0.0037	0.07
Kr-85m	Other Nobles	0.05	0.0066	0.13
Kr-87	Other Nobles	0.05	0.0034	0.07
Kr-88	Other Nobles	0.05	0.0047	0.09
Kr-89	Other Nobles	0.05	0.0007	0.01
Xe-131m	Other Nobles	0.05	0.0175	0.35
Xe-133m	Other Nobles	0.05	0.0114	0.23
Xe-133	Other Nobles	0.05	0.0150	0.30
Xe-135m	Other Nobles	0.05	0.0067	0.13
Xe-135	Other Nobles	0.05	0.0086	0.17
Xe-137	Other Nobles	0.05	0.0007	0.01
Xe-138	Other Nobles	0.05	0.0013	0.03
Rb-86	Alkali Metals	0.12	0.0195	0.16
Rb-88	Alkali Metals	0.12	0.0015	0.01
Rb-89	Alkali Metals	0.12	0.0014	0.01
Rb-90	Alkali Metals	0.12	0.0006	0.00
Cs-136	Alkali Metals	0.12	0.0253	0.21
Cs-138	Alkali Metals	0.12	0.0020	0.02
Cs-139	Alkali Metals	0.12	0.0011	0.01

3.1.4 Conclusions

A revised bounding operational power history has been evaluated for ONS fuel rods, with maximum linear heat generation rates exceeding 6.3 kW/ft for rod burnup above 54 GWD/MTU. This conservative rod power history, shown in Table 1, has been analyzed using an NRC-approved fuel performance code (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 method described in the ANS 5.4 [2011] standard.

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 for the long-lived Kr-85, Cs-134, and Cs-137 isotopes, as shown in Table 5. Computed fission gas release gap fractions for all other isotope groups remain well below the values from Table 3 of Regulatory Guide 1.183.

Table 5. Bounding Gap Fractions for Application to ONS Fuel Handling Accidents

Isotope or Isotope Group	Gap Fraction from Table 3 of Reg Guide 1.183 (Rev. 0)	Bounding Gap Fraction	Ratio
I-131	0.08	0.08	1
Kr-85	0.10	0.40	4
Other Noble Gases	0.05	0.05	1
Other Halogens	0.05	0.05	1
Cs-134 (Alkali Metal)	0.12	0.48	4
Cs-137 (Alkali Metal)	0.12	0.48	4
Other Alkali Metals	0.12	0.12	1

3.2 Fuel Handling Accident Dose Consequences

With the gap release analysis complete, the dose consequences were then determined for the ONS fuel handling accidents described in Section 2.0. It was assumed that all ONS HTP fuel rods involved in the accident could exceed the maximum linear heat generation rate limit of 6.3 kW/ft for burnups exceeding 54 GWD/MTU.

Based on the results provided in Table 5 (see Section 3.1.4), the gap release fractions for rods that exceed the 6.3 kW/ft LHGR limit above 54 GWD/MTU would increase only for Kr-85, Cs-134, and Cs-137, by a factor of 4, from Regulatory Guide 1.183. All other isotopes maintain the release fractions outlined in Table 3 of Regulatory Guide 1.183. For the fuel handling accidents, all radionuclide groups other than iodine and noble gases were assumed to remain in nonvolatile form, and not released from the pool.

Although some radionuclides have increased gap release fractions over the Regulatory Guide 1.183 values, the overall calculated dose results decreased from the current UFSAR values due to the use of only the ANS 5.4 [2011] standard that is based on more recent high burnup fuel data and has also shown the ANS 5.4 [1982] standard to be conservative (e.g., iodine gap release fractions in the [2011] standard remain as shown in Regulatory Guide 1.183). Table 7 shows the previous baseline doses prior to the Reference 10 submittal for comparison.

The source term used for the ONS fuel handling accidents in this LAR is unchanged from the current licensing basis source term (as presented in the Reference 10 LAR), and is provided again for convenience in Table 6.

Table 7 shows the dose consequences for the updated fuel handling accident evaluation, as a result of the changes to the Kr-85 gap fraction. The revised doses satisfy the requirements set forth in Regulatory Guide 1.183 and 10 CFR 50.67.

Table 6. Single Assembly Fuel Handling Accident Source Term for ONS

Isotope	ONS Source Term (Ci)
I-130	3.21E+04
I-131	6.82E+05
I-132	9.83E+05
I-133	1.38E+06
I-134	1.62E+06
I-135	1.31E+06
Kr-83m	1.10E+05
Kr-85m	2.53E+05
Kr-85	8.53E+03
Kr-87	5.18E+05
Kr-88	7.07E+05
Kr-89	9.02E+05
Xe-131m	1.02E+04
Xe-133m	4.27E+04
Xe-133	1.31E+06
Xe-135m	3.02E+05

Isotope	ONS Source Term (Ci)
Xe-135	4.26E+05
Xe-137	1.27E+06
Xe-138	1.30E+06
Rb-86	2.30E+03
Rb-88	7.11E+05
Rb-89	9.43E+05
Rb-90	8.89E+05
Cs-134	2.13E+05
Cs-136	6.05E+04
Cs-137	9.71E+04
Cs-138	1.38E+06
Cs-139	1.31E+06
Br-83	1.10E+05
Br-85	2.53E+05
Br-87	4.15E+05

Table 7. ONS Fuel Handling Accident Dose Consequences

Accident	Dose Results (Rem TEDE)			
	Baseline dose prior to Reference 10 submittal	Current UFSAR dose, after approval of Reference 10 submittal	**Revised Analysis (ANS 5.4 [2011] only)**	Dose Acceptance Criteria (Rem TEDE)
Fuel Handling Accident (Single Assembly Event)				
Exclusion Area Boundary (EAB)	1.18	1.33	**1.18**	6.3
Low Population Zone (LPZ)	0.13	0.14	**0.13**	6.3
Control Room	2.19	2.45	**2.19**	5.0
Fuel Cask Handling Accident (Multiple Assembly Event)				
Exclusion Area Boundary (EAB)	1.83	2.05	**1.93**	6.3
Low Population Zone (LPZ)	0.19	0.22	**0.21**	6.3
Control Room	3.61	4.05	**3.62**	5.0

4.0 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. 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 dose analyses, which is a change to the AST. These gap fraction calculations used a projected power history that bounds the current ONS 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 the fuel handling 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 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.

4.2 Precedent

The NRC has previously approved changes similar to the proposed changes in this LAR for several Duke Energy nuclear plants (Oconee, Catawba, and McGuire -- Reference 11; Robinson -- Reference 13; and Harris -- Reference 15). As with those LARs, ONS is proposing to follow the RG 1.183, Table 3, Footnote 11 alternative to calculate gap release fractions using bounding power histories and NRC-approved methodology. The analysis results similarly show that the radiological consequences of the fuel handling accidents remain within the regulatory dose acceptance criteria contained in RG 1.183, both for personnel offsite and operators in the control room. This submittal postulates fuel assemblies in future fuel cycle designs at ONS that have the potential to exceed the 6.3 kW/ft LHGR limit in RG 1.183, Table 3, Footnote 11 for burnups greater than 54 GWD/MTU.

4.3 No Significant Hazards Consideration Determination

Pursuant to 10 CFR 50.90, Duke Energy proposes a license amendment request (LAR) to change the Updated Final Safety Analysis Report (UFSAR) for Oconee Nuclear Station (ONS), Units 1, 2, and 3. Specifically, Duke Energy is requesting the Nuclear Regulatory Commission's (NRC'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 ONS reactor core design limits in order to support the request. Finally, the dose consequences contained in the ONS UFSAR for fuel handling 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?

   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 ONS. The dose consequences reported in the ONS Updated Final Safety Analysis Report (UFSAR) were reanalyzed for fuel handling 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 the fuel handling events at ONS meet the applicable TEDE dose acceptance criteria (specified also in RG 1.183).

   The changes proposed do not affect the precursors for fuel handling accidents analyzed in Chapter 15 of the ONS UFSAR. 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 previously evaluated?

   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 for certain isotopes, and were accounted for in the dose analysis for ONS. The dose consequences reported in the ONS UFSAR were reanalyzed for fuel handling 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 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 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?

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 for certain isotopes, and were accounted for in the dose analysis for ONS. The dose consequences reported in the ONS UFSAR were reanalyzed for fuel handling 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 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 LAR are the upper bounds to radiological consequences of the fuel handling 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 or 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 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.

5.0 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 a change to the amount of calculated radioactivity released, this change is not considered significant because the new dose consequences of the fuel handling-type accident 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.

6.0 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 (ML003716792).

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 (ML090960464).

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 (ML112070118).

6. NUREG/CR-7003, Background and Derivation of ANS 5.4 [2011] Standard Fission Product Release Model, J. Turnbull and C. Beyer, prepared for the U.S. Nuclear Regulatory Commission, January 2010 (ML100130186).

7. BAW-10231P-A, Revision 1, "COPERNIC Fuel Rod Design Computer Code," Framatome ANP (now AREVA), January 2004 (NP version - ML042930240).

8. NRC Information Notice IN 2009-23, "Nuclear Fuel Thermal Conductivity Degradation," October 2009 (ML091550527).

9. "Oconee Nuclear Station, Units 1, 2 and 3 RE: Issuance of Amendments (TAC Nos. MB3537, MB 3538, and MB3539)" -- letter from L. Olshan (U.S. NRC) to R. Jones (Duke Energy), June 1, 2004 (ML041540097).

10. "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" – submittal package from R. Repko (Duke Energy) to Nuclear Regulatory Commission, July 15, 2015 (ML15196A093).

11. "Catawba Nuclear Station, Units 1 and 2; McGuire Nuclear Station, Units 1 and 2; and Oconee Nuclear Station, Units 1, 2, and 3 - Issuance of Amendments Regarding Request to Use an Alternate Fission Gas Gap Release Fraction" – letter from J. Hall (U.S. NRC) to R. Repko (Duke Energy), July 19, 2016 (ML16159A336).

12. "H. B. Robinson Steam Electric Plant, Unit No. 2 - License Amendment Request to Modify the Licensing Basis Alternate Source Term" – submittal package from R. Glover (Duke Energy) to Nuclear Regulatory Commission, September 14, 2016 (ML16259A169).

13. "H. B. Robinson Steam Electric Plant, Unit No. 2 - Issuance of Amendment Regarding Request to Modify the Licensing Basis Alternate Source Term" – letter from D. Galvin (U.S. NRC) to E. Kapopoulos (Duke Energy), September 29, 2017 (ML17205A233).

14. "Shearon Harris Nuclear Power Plant, Unit 1 - 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" – submittal package from T. Hamilton (Duke Energy) to Nuclear Regulatory Commission, May 22, 2017 (ML17142A411).

15. "Shearon Harris Nuclear Power Plant, Unit 1 - Issuance of Amendment Regarding 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" – letter from M. Barillas (U.S. NRC) to T. Hamilton (Duke Energy), March 26, 2018 (ML18045A060).

ATTACHMENT

PROPOSED ONS UPDATED FINAL SAFETY ANALYSIS REPORT CHANGES

(MARK-UP)

15 pages plus cover page

4. Many of the transient and accident analyses involve control rod movement. These analyses credit the normal withdrawal sequence, overlap, and rod speed, which are controlled by non-safety control systems.

5. For certain failures in the EFW System, credit is taken for realigning EFW flow through the non-safety MFW System.

6. Steaming of the steam generators with manual non-safety atmospheric dump valves is credited.

7. Deleted per 2003 update

8. The capability to remotely throttle certain valves is credited. Some of the controls required to remotely throttle these valves are not safety-grade.

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.

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 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 (Reference 46 and 47). A maximum of 25 fuel rods, per fuel assembly, shall be allowed to exceed the rod power/burnup criteria for Footnote 11 in RG 1.183 in accordance with the license amendment request submitted by letter dated July 15, 2015 (Reference 46).~~ Insert A The fuel cycle design ensures that none of these fuel pins experience DNB following any design basis accident. 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

Insert A:

For non-DNB fuel pins that 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 4 for Kr-85, Cs-134, and Cs-137. The gap fractions for all other isotopes remain at their pertinent RG 1.183, Table 3 values (References 46 and 47).

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.

~~46. Repko, Regis T (Duke Energy) to USNRC, *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*, July 15, 2015.~~

~~47. Hall, James R (USNRC) to Repko, Regis T (Duke Energy), *Catawba Nuclear Station, Units 1 and 2; McGuire Nuclear Station, Units 1 and 2; Oconee Nuclear Station, Units 1, 2, and 3 - Issuance of Amendments Regarding Request to Use an Alternate Fission Gas Gap Release Fraction (CAC NOS. MF6480, MF6481, MF6482, MF6483, MF6484, MF6485, and MF6486)*, July 19, 2016.~~

Insert B

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46. This is a placeholder for the "Table 3 Footnote 11" (T3F11) 2018 license amendment request (LAR). This LAR will be identified upon NRC approval.

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.

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 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.~~ 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.

Insert C

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) = $73 e^{0.313 (t/d)}$

Insert C:

For fuel pins that 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 4 for Kr-85, Cs-134, and Cs-137. The gap fractions for all other isotopes remain at their pertinent RG 1.183, Table 3 values.

Where:

t = bubble rise time, seconds

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).

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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. ~~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.~~ The limiting doses for a fuel handling accident for a single fuel assembly event are given in Table 15-16.

Insert D

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.

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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.

Insert D:

For fuel pins that 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 4 for Kr-85, Cs-134, and Cs-137. The gap fractions for all other isotopes remain at their pertinent RG 1.183, Table 3 values.

10. The fractions of noble gases and iodine in the gaps are shown below. ~~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 (Reference 1).~~

Insert E

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

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The offsite radiological consequences of the postulated cask drop accident in either spent fuel pool is within the Regulatory Guide 1.183 limits. The limiting doses for a fuel cask handling accident for a multiple fuel assembly event are given in Table 15-16.

15.11.2.5 Dry Storage Transfer Cask Drop Accident in Spent Fuel Pool Building

Dry storage transfer operations from the spent fuel pool (SFP) buildings to the Independent Spent Fuel Storage Facility (ISFSI) are routinely performed at Oconee. The major steps in the process involve transporting the transfer cask/dry storage canister (DSC) into the fuel building, placing into the SFP, loading with 24 qualified fuel assemblies, drying/sealing, and removing to the ISFSI. The potential exists for dropping the cask in the SFP area during transfer operations.

15.11.2.5.1 Criticality Analyses for Dry Storage Transfer Cask Drop Scenarios

While the transfer cask is never carried directly over spent fuel, the potential always exists for failure of the overhead crane or handling equipment. Thus, an analysis was performed assuming the cask, yoke, and yoke block are deflected into the Unit 1&2 SFP. In such a case, it was postulated that 1024 spent fuel assemblies (SFAs) would be damaged (the first 64 rows, each containing 16 SFAs). It was assumed that 220 fuel storage cells directly beneath the falling parts buckle and deflect into adjacent cells until all the energy of the dropping cask is absorbed. For a cask drop in the smaller Unit 3 SFP, it was assumed all 825 fuel cell locations would be damaged.

The potential for criticality in the SFPs was analyzed using the methodology identified in NUREG0612. It was assumed the racks and fuel were deformed such that keff was maximized. Credit was taken for pool boron and stainless steel walls to determine the keff under the assumed damage conditions. The confirmatory calculations utilized a specific neutronic analysis for each SFP with the following assumptions:

1. An infinite array of SFAs is crushed together into a geometry that optimizes keff.
2. The affected SFAs are unirradiated and have the maximum enrichment permitted for storage in the Oconee SFPs.
3. The minimum technical specification for SFP boron concentration is maintained.

The acceptance criteria for this accident per NUREG0612, is that k_{eff} will be less than or equal to 0.95 including all uncertainties. A series of calculations involving cases of varied pin pitch modeling the crushed cells and SFAs was performed. The maximum k_{eff} value determined for

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For fuel pins that 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 4 for Kr-85, Cs-134, and Cs-137. The gap fractions for all other isotopes remain at their pertinent RG 1.183, Table 3 values (Reference 1).

15.11.3 References

1. DPC Engineering Calculation OSC7738, "Fuel Handling Accidents (FHA) Dose Analysis", dated ~~January 28, 2010.~~ September 10, 2018.
2. WCAP7828, "Radiological Consequences of a Fuel Handling Accident", December 1971.
3. DPC Engineering Calculation OSC9154, "Oconee Isotopic Source Term Calculations", dated July 23, 2008.
4. Deleted per 2005 Update.
5. Parker, W. O. Jr., Letter to Rusche, B. C. (NRC), November 3, 1975.
6. Parker, W. O. Jr. (Duke), Letter to Denton, H. R. (NRC), July 25, 1980.
7. Tucker, H. B. (Duke), Letter to Denton, H. R. (NRC), November 19, 1985.
8. Deleted per 2005 Update.
9. DPC Engineering Calculation OSC3631 Rev 2, "Criticality Consequences of a Heavy Load Drop in the Spent Fuel Pool", dated February 7, 1996.
10. Deleted per 2002 update.
11. Oconee Nuclear Station Site Specific Independent Spent Fuel Storage Installation, Final Safety Analysis Report, Chapter 8.
12. Oconee Nuclear Station Independent Spent Fuel Storage Installation General License, Updated Final Safety Analysis Report.
13. Deleted per 2006 update.
14. Deleted per 2006 update.
15. Deleted per 2006 update.
16. Deleted per 2006 update.

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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 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.~~

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For fuel pins that 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 4 for Kr-85, Cs-134, and Cs-137. The gap fractions for all other isotopes remain at their pertinent RG 1.183, Table 3 values.

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	~~1.33~~ 1.18	
TEDE at LPZ	~~0.14~~ 0.13	
TEDE in Control Room	~~2.45~~ 2.19	
Deleted row(s) per 2016 update		
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	~~2.05~~ 1.93	
TEDE at LPZ	~~0.22~~ 0.21	
TEDE in Control Room	~~4.05~~ 3.62	
Deleted row(s) per 2008 update		
Deleted row(s) per 2004 update		
Deleted row(s) per 2008 update		