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10 CFR 50.90

May 22, 2017
Serial: HNP-17-033

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

Shearon Harris Nuclear Power Plant, Unit 1
Docket No. 50-400/Renewed License No. NPF-63

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:

In accordance with the provisions of 10 CFR 50.90, Duke Energy Progress, LLC (Duke Energy), hereby submits a license amendment request for Shearon Harris Nuclear Power Plant, Unit 1 (HNP). This amendment would revise the facility as described in the Final Safety Analysis Report (FSAR) 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 HNP, and submitted herein, support an increase to the Reference 1 LHGR limit.

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 the fuel handling accidents at HNP 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 FSAR pages are included in 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.

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.

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

This document contains no new Regulatory Commitments.

Please refer any questions regarding this submittal to Jeff Robertson, HNP Regulatory Affairs Manager, at (919) 362-3137.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on May 22, 2017.

Sincerely,

Tanya M. Hamilton

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

cc: J. Zeiler, NRC Sr. Resident Inspector, HNP
W. L. Cox, III, Section Chief N.C. DHSR
M. Barillas, NRC Project Manager, HNP
C. Haney, NRC Regional Administrator, Region II

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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

1.0 SUMMARY DESCRIPTION

This technical evaluation supports a request to amend the Operating License NPF-63 for Shearon Harris Nuclear Power Plant, Unit 1 (HNP).

The proposed changes would revise the dose consequences for the facility, as described in the HNP Final Safety Analysis Report (FSAR), 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:

> "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."

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

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

These increased gap fractions allow LHGRs above 6.3 kW/ft for rod burnup above 54 GWD/MTU, as long as the LHGRs remain within the bounding power history evaluated in Section 3.1.

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 method in the ANS 5.4 [2011] standard (Reference 4). Duke Energy has previously submitted LARs related to increasing high-burnup LHGRs for Catawba Nuclear Station, McGuire Nuclear Station, and Oconee Nuclear Station (Reference 11 – LAR; Reference 12 – Amendment Issuance), and for H.B. Robinson Steam Electric Plant (Reference 13 – LAR).

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 17x17 assembly type currently operating in the HNP reactor.

Section 3.2 includes an evaluation of the dose consequences of a fuel handling accident in Containment or in the Fuel Handling Building, in which the damaged fuel assemblies include high-burnup fuel pins operated above 6.3 kW/ft. No non-LOCA accidents that may result in departure from nucleate boiling (DNB) are considered (e.g., locked rotor accident, rod ejection

accident, single RCCA withdrawal, inadvertent fuel assembly core loading error, main steam line break, etc). Fuel cycles for HNP 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 changes proposed in this LAR would be reflected in updates to Section 15.7.4 of the HNP FSAR, which addresses design basis fuel handling accidents. Draft markups to the HNP FSAR are provided in Enclosure 2.

3.0 TECHNICAL EVALUATION

Gap release fractions for high-burnup rods (greater than 54 GWD/MTU) with an increased 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 accidents in Containment or in the Fuel Handling Building. The HNP 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.

HNP 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, 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.

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 AREVA HTP 17x17 fuel rod design is considered for the fission gas release calculations. This is the design that is currently being irradiated in the HNP reactor. As this fuel type is representative of a general 17x17 pressurized water reactor (PWR) design, the analysis of the fuel rod is judged to be applicable to other 17x17 designs that may be used in the HNP reactor.

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 HNP 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, consistent with the restrictions of the ANS 5.4 [2011] method.

The HNP core average power (deposited within the fuel rod) is calculated below. The 0.974 value represents the fraction of total heat from fission that is deposited within the fuel rod. The 1.0034 factor accounts for the uncertainty in measured core power (Reference 10).

<u>HNP:</u>

$$avg\ rod\ power = \left(\frac{2948000\ kW_{th} \times 1.0034 \times 0.974}{157\ assys \times 264 \frac{rods}{assy} \times 12 \frac{ft}{rod}} \right) = 5.793\ \frac{kW}{ft}$$

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

HNP HTP fuel	
Rod Burnup Range (GWD/MTU)	**Average Rod Power (kW/ft)**
0 – 2	9.600
2 – 4	9.600
4 – 6	9.600
6 – 8	9.600
8 – 10	9.600
10 – 12	9.600
12 – 14	9.600
14 – 16	9.600
16 – 18	9.600
18 – 20	9.600
20 – 22	9.600
22 – 24	9.600
24 – 26	9.532*
26 – 28	9.395*
28 – 30	9.258*
30 – 32	9.121*
32 – 34	8.984*
34 – 36	8.847*
36 – 38	8.711*
38 – 40	8.574*
40 – 42	8.437*
42 – 44	8.300*
44 – 46	8.163*
46 – 48	8.026*
48 – 50	7.889*
50 – 52	7.753*
52 – 54	7.616*
54 – 56	7.479*
56 – 58	7.342*
58 – 60	7.205*
60 – 62	7.068*

* Values shown are at the midpoint of the pertinent burnup interval. The bounding LHGR from 24 to 62 GWD/MTU is a linear function: LHGR = 11.242 - 0.06842*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:

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

where:

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

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

$\left(\frac{S}{V}\right)_{i,m}$ 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:

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

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

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})

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

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

$Diam_o$ 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 HNP 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 (~ 0.974 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 a recent HNP 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 HNP UO_2 fuel with no integral gadolinia poisons. Previous analysis (Reference 11) 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 a HNP 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 were obtained from a recent HNP core design, as a function of rod burnup. Using these shapes and the other input information detailed above, a COPERNIC case was executed for a HNP HTP fuel rod with a 5.00 wt % U-235 central fuel enrichment and 2.60 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 11 and Reference 13 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 3 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.

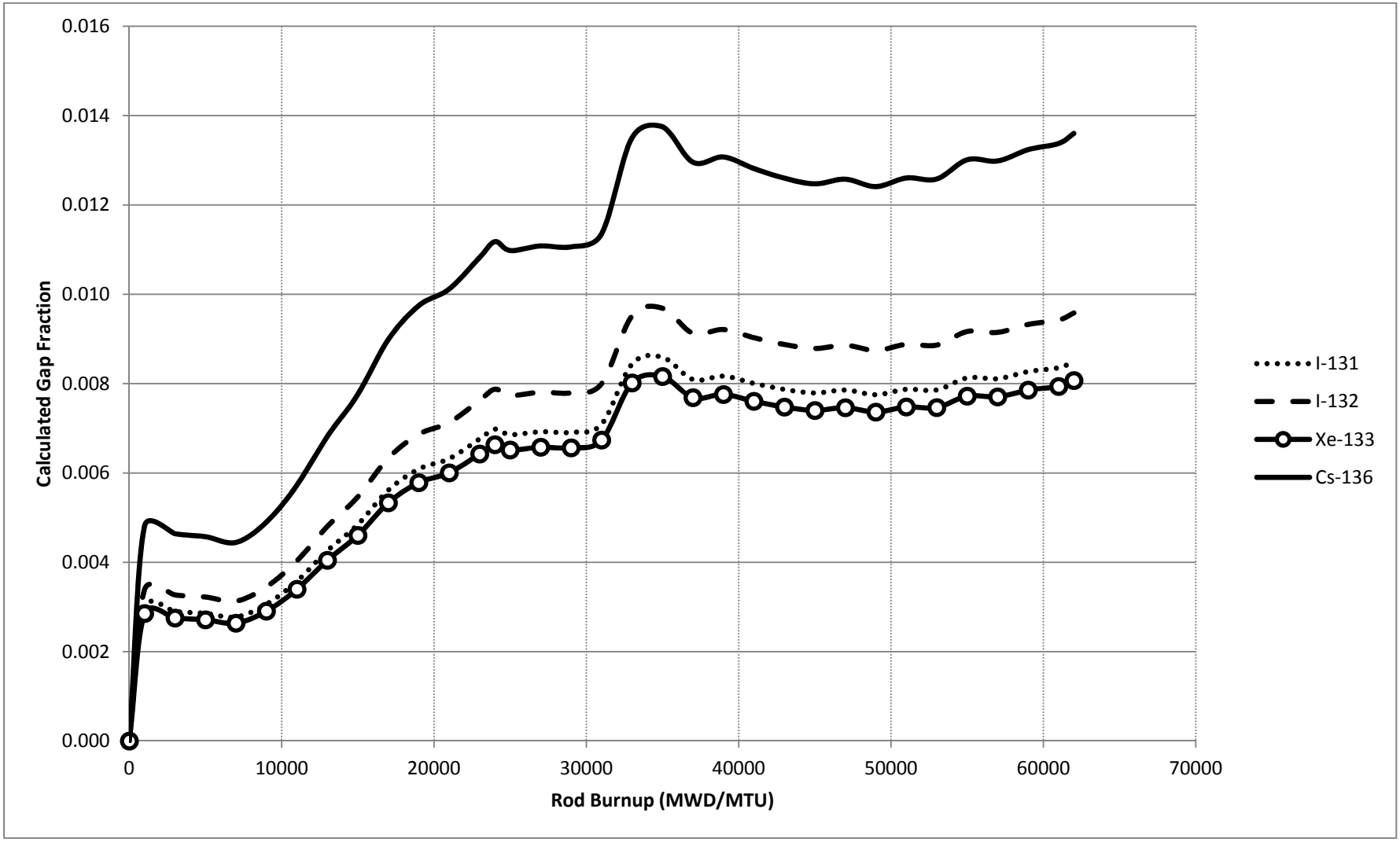

Figure 1. Calculated Gap Fractions for Selected Short-Lived Isotopes – 5.00 wt % U-235 HNP HTP Fuel

Table 4. Results from Gap Release Calculations

Isotope	Isotope Category	Reg Guide 1.183 Table 3 Value	HNP HTP fuel calculated maximum gap fraction		Ratio
Long-lived (> 1-yr half-life) Isotopes (from COPERNIC results)					
Kr-85	Kr-85	0.10	0.212		2.12
Cs-134	Alkali Metals	0.12	0.300		2.50
Cs-137	Alkali Metals	0.12	0.300		2.50
Short-lived (< 1-yr half-life) Isotopes (from gapfrac results)					
I-130	Other Halogens	0.05	0.0043		0.09
I-131	I-131	0.08	0.0086		0.11
I-132	Other Halogens	0.05	0.0097		0.19
I-133	Other Halogens	0.05	0.0052		0.10
I-134	Other Halogens	0.05	0.0029		0.06
I-135	Other Halogens	0.05	0.0037		0.07
Br-83	Other Halogens	0.05	0.0023		0.05
Br-85	Other Halogens	0.05	0.0003		0.01
Br-87	Other Halogens	0.05	0.0002		0.00
Kr-83m	Other Nobles	0.05	0.0020		0.04
Kr-85m	Other Nobles	0.05	0.0036		0.07
Kr-87	Other Nobles	0.05	0.0019		0.04
Kr-88	Other Nobles	0.05	0.0025		0.05
Kr-89	Other Nobles	0.05	0.0004		0.01
Xe-131m	Other Nobles	0.05	0.0095		0.19
Xe-133m	Other Nobles	0.05	0.0062		0.12
Xe-133	Other Nobles	0.05	0.0082		0.16
Xe-135m	Other Nobles	0.05	0.0036		0.07
Xe-135	Other Nobles	0.05	0.0047		0.09
Xe-137	Other Nobles	0.05	0.0004		0.01
Xe-138	Other Nobles	0.05	0.0007		0.01
Rb-86	Alkali Metals	0.12	0.0106		0.09
Rb-88	Alkali Metals	0.12	0.0008		0.01
Rb-89	Alkali Metals	0.12	0.0007		0.01
Rb-90	Alkali Metals	0.12	0.0003		0.00
Cs-136	Alkali Metals	0.12	0.0137		0.11
Cs-138	Alkali Metals	0.12	0.0011		0.01
Cs-139	Alkali Metals	0.12	0.0006		0.00

3.1.4 Conclusions

A bounding operational power history has been evaluated for HNP 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 HNP 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	0.30	3
Other Noble Gases	0.05	0.05	1
Other Halogens	0.05	0.05	1
Cs-134 (Alkali Metal)	0.12	0.36	3
Cs-137 (Alkali Metal)	0.12	0.36	3
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 HNP fuel handling accidents in Containment and the Fuel Handling Building. It was assumed that all HNP 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, which are tripled 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 are assumed to remain in nonvolatile form, and not released from the pool.

The source term used for the HNP fuel handling accidents in this LAR is unchanged from the current licensing basis source term (Reference 9). It accounts for PWR assembly operation with a constant peaking factor of 1.73 and also assumes a boiling water reactor (BWR) operation with a constant peaking factor of 1.50 for any affected BWR assemblies involved in the accident [note that discharged Brunswick BWR fuel assemblies are currently being stored in the HNP spent fuel pools]. Table 6 provides a listing of the pertinent HNP fuel handling accident input parameters. The Control Room unfiltered inleakage has been decreased from 500 cfm to 300 cfm for consistency with the value used for the LBLOCA accident (as noted in the HNP FSAR). This unfiltered inleakage change lowers the calculated Control Room doses and more than offsets any dose increase associated with the tripling of the Kr-85 gap release fractions.

In addition, the computer code used for the updated HNP fuel handling accident dose analysis was changed, from Westinghouse TITAN5 to Bechtel LOCADOSE. The LOCADOSE code has previously been used by Duke Energy in accident dose analyses that supported AST licensing submittals (e.g., Reference 14).

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, Control Room inleakage, and the dose analysis computer code, the updated HNP maximum fuel handling accident doses (associated with an accident in Containment) are 2.02 Rem TEDE [Total Effective Dose Equivalent] for the Exclusion Area Boundary (EAB), 0.46 Rem TEDE for the Low Population Zone (LPZ), and 0.88 Rem TEDE for the Control Room. The revised doses satisfy the requirements set forth in Regulatory Guide 1.183 and 10 CFR 50.67.

Table 6. HNP Fuel Handling Accident - Dose Consequence Analysis Inputs

Input or Assumption Description	Current Licensing Basis Value	Updated Value
Reactor power with uncertainty (MW)	2958	No change
Source Term (Ci)	**FSAR Tables 15.7.4-1 and 15.7.4-3**	**See Changes in Markups provided in Enclosure 2**
Number of fuel rods damaged	264 (Containment) or 314 + 52 BWR assemblies (Fuel Handling Bldg)	No change
Peaking factor	1.73 (PWR assemblies) and 1.50 (BWR assemblies)	No change
Decay time (hr)	100 hours (PWR assemblies) and 4 years (BWR assemblies)	No change
Number of high-burnup (> 54 GWD/MTU) rods that can exceed 6.3 kW/ft	**0**	**264 (Containment) or 314 (Fuel Handling Bldg)**
Kr-85 release fraction for high-burnup rods above 6.3 kW/ft	**Not permitted**	**0.30**
Control room volume (ft^3)	71,000	No change
Control room ventilation Iodine removal efficiencies (elemental, particulate, organic %)	(99, 99, 99)	No change
Control room ventilation unfiltered inleakage (cfm)	**500**	**300**
Fuel Pool Decontamination Factor - Iodine	200	No change
Fuel Pool Decontamination Factor - Noble Gases	1	No change
Depth of water above fuel inside Containment (ft)	22	No change
Depth of water above fuel inside Fuel Handling Building (ft)	21	No change
Duration of Release (hours)	2	No change
Fuel Handling Building Iodine filter efficiencies % (elemental, organic, particulate)	(95, 95, 95)	No change
Containment Iodine filter efficiencies % (elemental, organic, particulate)	No Filtration Assumed	No change

Table 7. HNP Fuel Handling Accident Dose Consequences

Accident	**Dose Results (Rem TEDE)**		
	Current FSAR	**Updated Analysis**	Dose Acceptance Criteria (Rem TEDE)
Fuel Handling Accident -- Containment			
Exclusion Area Boundary (EAB)	2.03	**2.02**	6.3
Low Population Zone (LPZ)	0.46	**0.46**	6.3
Control Room	1.39	**0.88**	5.0
Fuel Handling Accident -- Fuel Handling Building			
Exclusion Area Boundary (EAB)	0.34	**0.35**	6.3
Low Population Zone (LPZ)	0.077	**0.078**	6.3
Control Room	0.12	**0.083**	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 HNP. 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 HNP dose analyses, which is a change to the AST. These gap fraction calculations used a projected power history that bounds the current HNP 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 HNP 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 other nuclear power plants including Catawba Nuclear Station, McGuire Nuclear Station, and Oconee Nuclear Station (Reference 12). Similar to these plants, HNP 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. Additionally, the analysis results similarly show 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 also postulates fuel assemblies in future fuel cycle designs at HNP 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 Progress, LLC (Duke Energy), proposes a license amendment request (LAR) to change the Final Safety Analysis Report (FSAR) for Shearon Harris Nuclear Power Plant, Unit 1 (HNP). 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 HNP reactor core design limits in order to support the request. Finally, the dose consequences contained in the HNP FSAR 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 HNP. The dose consequences reported in the Final Safety Analysis Report (FSAR) 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 HNP 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 accidents analyzed in Chapter 15 of the HNP FSAR. The probability remains unchanged since the accident analyses performed and discussed in the basis for the FSAR changes involve no change to a system, structure or component that affects initiating events for any FSAR 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 and accounted for in the dose analysis for HNP. The dose consequences reported in HNP's FSAR 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 HNP. 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 HNP 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 and accounted for in the dose analysis for HNP. The dose consequences reported in HNP's FSAR 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 HNP. 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 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.

6.0 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 (ADAMS Accession No. 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 (ADAMS Accession No. 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 (ADAMS Accession No. 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 (ADAMS Accession No. ML100130186).

7. BAW-10231P-A, Revision 1, "COPERNIC Fuel Rod Design Computer Code," Framatome ANP (now AREVA), January 2004 (NP version - ADAMS Accession No. ML042930240).

8. NRC Information Notice IN 2009-23, "Nuclear Fuel Thermal Conductivity Degradation," October 2009 (ADAMS Accession No. ML091550527).

9. "Shearon Harris Nuclear Power Plant, Unit 1 - Issuance of Amendment Re: Steam Generator Replacement and Power Uprate" – letter from N. Kalyanam (U.S. NRC) to J. Scarola (Carolina Power & Light Company), October 12, 2001 (ADAMS Accession No. ML012830516).

10. "Shearon Harris Nuclear Power Plant, Unit 1 - Issuance of Amendment Re: Measurement Uncertainty Recapture Power Uprate" – letter from A. Colón (U.S. NRC) to C. Burton (Progress Energy Carolinas), May 30, 2012 (ADAMS Accession No. ML11356A096).

11. "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 (ADAMS Accession No. ML15196A093).

12. "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 (ADAMS Accession No. ML16159A336).

13. "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 (ADAMS Accession No. ML16259A169).

14. "McGuire Nuclear Station, Units 1 and 2, Issuance of Amendments Regarding Adoption of the Alternate Source Term Radiological Analysis Methodology" – letter from J. Stang (U.S. NRC) to B. Hamilton (Duke Energy), March 31, 2009 (ADAMS Accession No. ML090890627).

ENCLOSURE 2

PROPOSED FINAL SAFETY ANALYSIS REPORT CHANGES (MARK-UP)

9 pages plus cover page

15.7.4 Design Basis Fuel Handling Accidents

15.7.4.1 Identification of Causes and Accident Description. The possibility of a fuel handling accident is remote because of the many interlocks, administrative controls, and physical limitations imposed on the fuel handling operations. All refueling operations are conducted in accordance with prescribed procedures under direct surveillance of a senior reactor operator (SRO). The analyzed Fuel Handling Accident inside containment involves dropping a spent fuel assembly resulting in the rupture of the cladding of all the fuel rods (264) in the assembly.

The projected worst case Fuel Handling Accident (FHA) in the Fuel Handling Building (FHB) involves dropping a recently discharged (100 hr decayed) PWR assembly (including the handling tool) on top of another recently discharged PWR assembly in a fuel storage rack. The dropped assembly subsequently falls over landing on BWR fuel assemblies in an adjacent storage rack. Fifty fuel rods are projected to fail in the impacted PWR assembly in storage and all of the rods (264) in the dropped assembly fail when the assembly falls over (Reference 15.7.4.5). Due to the upper bail handle of the BWR fuel assemblies extending above the top of the BWR storage racks, up to 52 BWR assemblies could be impacted when the dropped PWR assembly falls over. All of the rods in the impacted BWR assemblies are assumed to fail.

15.7.4.2 Radiological Consequences Analysis

15.7.4.2.1 Input Assumptions Common to both FHA in the FHB and in Containment.

Consistent with Regulatory Guide 1.183 (Position 1.2 of Appendix B), the radionuclides considered are xenons, kryptons, halogens, cesiums and rubidiums. The list of xenons, kryptons, and halogens considered is given in Tables 15.7.4-1 and 15.7.4-3. The cesium and rubidium are not included because they are not assumed to be released from the pool as discussed later.

The calculation of the radiological consequences following a FHA uses gap fractions of 8% for I-131, ~~10%~~ 30% for Kr-85, and 5% for all other pertinent nuclides (Reference 15.7.4-11).

Iodine species in the pool is 99.85% elemental and 0.15% organic iodine. This is based on the split leaving the fuel of 95% cesium iodide (CsI), 4.85% elemental iodine and 0.15% organic iodine. It is assumed that all CsI is dissociated in the water and re-evolves as elemental. This is assumed to occur instantaneously. Thus, 99.85% of the iodine released is elemental.

The water above the damaged fuel rods retains a large fraction of the gap activity of iodines. An overall effective decontamination factor (DF) of 200 is used. The split between elemental and organic iodine leaving the pool has no impact on the analyses since the filter efficiencies credited in the analyses for the two forms of iodine are the same.

The cesium and rubidium released from the damaged fuel rods is assumed to remain in a nonvolatile form and would not be released from the pool.

15.7.4.2.2 Postulated Fuel Handling Accident in the FHB

The major assumptions and parameters used in the analysis are itemized in Table 15.7.4-1. This analysis involves dropping a recently discharged (100 hour decay) PWR fuel assembly onto 52 Brunswick BWR fuel assemblies. This analysis also includes 50 PWR rods additionally damaged in the accident. the assembly inventory is based on the assumption that the PWR fuel assembly has been operated at 1.73 times the core average power and the BWR fuel assemblies have been operated at 1.5 times the core average power. All activity released from the fuel pool is assumed to be released to the atmosphere in two hours.

The BWR fuel inventory was conservatively evaluated at the IF-300 spent fuel shipping cask limits for GE- 7, 8, 9, 10, and 13 fuel assemblies with a maximum average lattice enrichment of 4.25 wt. % U-235 and a maximum assembly average burnup of 45 GWD/MTU. The decay time used in the analysis is 100 hours for the PWR fuel and 4 years for the BWR fuel. Thus, the analysis supports the design basis limit of 100 hours decay time prior to fuel movement.

It was determined that for the HNP specific water height above the failed fuel in the fuel handling building of 21 feet, the elemental DF would be at least 291, compared to the Reg. Guide 1.183 allowable elemental DF of 500. Using the elemental DF 291, it was determined that overall effective DF for 21 feet of coverage would be 203. Since this continues to exceed the Reg. Guide 1.183 cited overall effective DF of 200, it remains conservative to use the overall DF of 200 in the HNP dose calculations.

Credit is taken for removal of iodine by filters by the spent fuel pool ventilation system operation. Credit is not taken for isolation of release paths.

The activity released from the damaged assemblies is assumed to be released to the fuel building and subsequently to the atmosphere over a 2 hour period.

15.7.4.2.3 Postulated Fuel Handling Accident in Containment

A fuel assembly is assumed to be dropped in containment and damaged during refueling. Activity released from the damaged assembly is released to the outside atmosphere through the containment openings (such as the personnel air lock door or the equipment hatch).

The major assumptions are parameters used in the analysis are itemized in Table 15.7.4-3. This analysis involves dropping a recently discharged (100 hour decay) PWR fuel assembly. All activity released from the fuel pool is assumed to be released to the atmosphere in two hours. The pool referred to in RG 1.183 is interpreted as the flooded reactor cavity for the purposes of evaluating the fuel handling accident in containment. No credit is taken for isolation of containment for the FHA containment.

The calculation of the radiological consequences following a FHA uses gap fractions of 8% for I-131, ~~10%~~ 30% for Kr-85, and 5% for all other pertinent nuclides (Reference 15.7.4-11).

It is assumed that all of the fuel rods in the equivalent of one fuel assembly (264 rods) are damaged to the extent that all their gap activity is released. The assembly inventory is based on the assumptions that the subject fuel assembly has been operated at 1.73 times the core average power.

The decay time used in the analysis is 100 hours.

It was determined that for HNP specific water height above the failed fuel in the containment of 22 feet, the elemental DF would be at least 382, compared to the Reg. Guide 1.183 allowable elemental DF of 500. Using the elemental DF of 382, it was determined that the overall effective DF for 22 feet of coverage would be 243. Since this continues to exceed the Reg. Guide 1.183 cited overall effective DF of 200, it remains conservative to use the overall DF of 200 in the HNP dose calculations.

No credit is taken for removal of iodine by filters nor is credit taken for isolation of release paths.

Although the containment purge will be automatically isolated on a purge line high radiation alarm, isolation is not modeled in the analysis. The activity released from the damaged assembly is assumed to be released to the outside atmosphere over a 2 hour period. Since no filters or containment isolation is modeled, this analysis supports refueling operation with the equipment hatch or personnel air lock remaining open.

15.7.4.2.4 Offsite Doses

The offsite doses are calculated using the assumptions and equations in Section 15.0A.1.

~~15.7.2.2.5~~ 15.7.4.2.5 Control Room Doses

The control room assumptions are provided in Section 15.6.5.4.3 and table 15.6.5-15. The FHA control room doses modeled ~~500~~ 300 cfm unfiltered inleakage.

It is assumed that the control room HVAC system begins in normal mode. The activity level in the intake duct causes a high radiation signal almost immediately. It is conservatively assumed that the post accident recirculation control room HVAC mode is entered 15 seconds after event initiation. The control room HVAC is placed into pressurization mode at 2 hours after isolation signal.

15.7.4.2.6 Results

The analytically predicted dose consequence to the Control Room (CR) operators due to LBLOCA increased by a small amount, based on the reduction in CR recirculation flow which reduces the iodine filtration provided by the charcoal filters relative to the analysis of record condition. The analytically predicted dose consequence to the control room operators for the FHA events were not similarly revised or updated since the LBLOCA DBA is the limiting control room operator dose event. The radiological analysis results for the FHA in FHB doses are listed in Table 15.7.4-2. The FHA in Containment doses are listed in Table 15.7.4-4. The TEDE doses have been analyzed for the worst two hours at the EAB and for the duration of the event at the LPZ and in the control room. The resultant doses are within the applicable limits. The offsite doses are less than ~25% of the 10CFR50.67 limits (i.e., 6.3 rem TEDE) and the control room dose is less than the 10CFR50.67 limit of 5 rem TEDE.

15.7.4.3 DELETED

15.7.4.3.1 DELETED

15.7.4.3.2 DELETED

15.7.4.4 Deleted.

15.7.4.4.1 Deleted.

15.7.4.4.2 Deleted.

15.7.4.4.3 Deleted.

15.7.4.4.4 Deleted.

15.7.4.5 Other Fuel Handling Accidents. Fuel handling drop accidents involving the other fuel handling tools (BPRA, RCCA change tool, spent fuel handling tool), and items carried by the tools have also been evaluated (Reference 15.7.4-7) and are addressed in Section 9.1. The tool drop scenarios involve dropping the tools, and items carried by the tools, onto PWR spent fuel racks, BWR spent fuel racks, and combinations of both. For all cases evaluated, the off-site dose consequences were determined to be bounded by the Fuel Handling Accident described in FSAR Section ~~15.7.4.5~~ 15.7.4.2.2 which addresses a fuel handling drop accident which results in damage to 314 PWR spent fuel rods and 52 BWR spent fuel assemblies (Reference 15.7.4-7, pages 3.2.2-3.23).

TABLE 15.7.4-1

Parameters Used in Fuel Handling Accident Inside the Fuel Handling Building Radiological Analysis

Radial peaking factor (PWR fuel)	1.73
(BWR fuel)	1.5
INSERT Fuel damaged (number of assemblies)	1.2 PWR (314 rods) + 52 BWR
↳ [NOTE: All damaged PWR rods assumed to exceed 6.3 kW/ft above 54 GWD/MTU burnup]	
Time from shutdown before fuel movement (PWR)(hr)	100
(BWR fuel)(yr)	4
Activity in the damaged fuel assembies (Ci)	
I-131	7.21E5
I-133	7.59E4
I-135	5.57E1
Kr-85	1.41E5
Xe-131m	9.06E3
Xe-133m	1.77E4
Xe-133	1.19E6
Xe-135	2.41E2
Gap Fractions (% of core activity)	
I-131	8
Kr-85	~~10~~ 30
Other Iodine and Noble Gas nuclides	5
Water depth	21 feet
Overall pool iodine scrubbing factor	200
Iodine chemical form in release to atmosphere (%)	
Elemental	70
Organic	30
Particulate	0
Spent Fuel Pool Ventilation System Filter efficiency	
Elemental	95
Organic	95
Particulate	95
Isolation of release	No isolation assumed
Time to release all activity (hours)	2
Activity Released, (Ci)	
I-131	~~1.439E1~~ 1.442E1
I-133	~~9.471E-1~~ 9.488E-1
I-135	~~6.950E-4~~ 6.963E-4
Kr-85	~~1.410E4~~ 4.230E4
Xe-131m	4.530E2
Xe-133m	8.850E2
Xe-133	5.950E4
Xe-135	1.205E1

TABLE 15.7.4-2

RADIOLOGICAL CONSEQUENCES OF A POSTULATED FUEL HANDLING ACCIDENT IN THE FUEL HANDLING BUILDING

Exclusion Area Boundary*	0.35 ~~0.34~~ rem TEDE
Low Population Zone	0.078 ~~0.077~~ rem TEDE
Control Room	0.083 ~~0.12~~ rem TEDE

* The exclusion area boundary dose reported is for the worst two hour period. determined to be from 0 to 2 hours.

TABLE 15.7.4-3

Parameters Used in a Fuel Handling Accident Inside Containment Radiological Analysis

Radial peaking factor	1.73
Fuel damaged (number of assemblies)	1
Time from shutdown before fuel movement (hr)	100
Activity in the damaged fuel assembly (Ci)	
I-131	6.06E5
I-133	6.38E4
I-135	4.68E1
Kr-85	8.82E3
Xe-131m	7.61E3
Xe-133m	1.49E4
Xe-133	9.97E5
Xe-135	2.03E2
Gap Fractions (% of core activity)	
I-131	8
Kr-85	~~10~~ 30
Other Iodine and Noble Gas nuclides	5
Water depth	22 feet
Overall pool iodine scrubbing factor	200
Iodine chemical form in release to atmosphere (%)	
Elemental	70
Organic	30
Particulate	0
Filter efficiency	No filtration assumed
Isolation of release	No isolation assumed
Time to release all activity (hours)	2
Activity Released (Ci)	
I-131	~~2.420E2~~ 2.424E2
I-133	~~1.592E1~~ 1.595E1
I-135	~~1.168E-2~~ 1.170E-2
Kr-85	~~8.820E2~~ 2.646E3
Xe-131m	3.805E2
Xe-133m	7.450E2
Xe-133	4.985E4
Xe-135	1.015E1

INSERT

[NOTE: All damaged fuel rods assumed to exceed 6.3 kW/ft above 54 GWD/MTU burnup]

TABLE 15.7.4-4

Radiological Consequences of a Postulated Fuel Handling Accident Inside Containment

Exclusion Area Boundary*	2.02 ~~2.03~~ rem TEDE
Low Population Zone	0.46 rem TEDE
Control Room	0.88 ~~1.39~~ rem TEDE

* The exclusion area boundary dose reported is for the worst two hour period, determined to be from 0 to 2 hours.

REFERENCES: SECTION 15.7

15.7.4-1 Industrial Ventilation, 8th Edition, American Conference of Governmental Industrial Hygienists.

15.7.4-2 Deleted by Amendment No. 49

15.7.4-3 Deleted by Amednment No. 51

15.7.4-4 Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors", July 2000.

15.7.4-5 Westinghouse Letter, 97CP-G-0006; Christine M. Vertes to Leo Martin, dated April 9, 1997, "Limiting Fuel Handling Accident Assumptions."

15.7.4-6 Deleted by Amendment No. 49

15.7.4-7 ESR 98-00181 "Fuel Handling Tool Drop onto Spent Fuel Rack Evaluation"

15.7.4-8 Deleted by Amendment No. 51

15.7.4-9 Deleted by Amendment No. 51

7 15.7.4-10 CP&L Calculation HNP-M/FHB-1001 "Off-site Doses from FHB Cask Drop."

INSERT:

15.7.4-11 Reference for pending HNP RG 1.183 Table 3 Footnote 11 SER