UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

September 29, 2017

Mr. Ernest J. Kapopoulos, Jr.
Site Vice President
H. B. Robinson Steam Electric Plant
Duke Energy Progress, LLC
3581 West Entrance Road, RNPA01
Hartsville, SC 29550

SUBJECT: H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2 – ISSUANCE OF AMENDMENT REGARDING REQUEST TO MODIFY THE LICENSING BASIS ALTERNATE SOURCE TERM (CAC NO. MF8378)

Dear Mr. Kapopoulos:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 255 to Renewed Facility Operating License No. DPR-23 for the H. B. Robinson Steam Electric Plant, Unit No. 2 (Robinson). This amendment is being issued in response to your application dated September 14, 2016.

The amendment authorizes the modification of the Robinson Updated Final Safety Analysis Report to reflect the adoption of a revised alternative source term to support the transition from an 18-month to a 24-month fuel cycle. The amendment also authorizes the inclusion of a related dose consequence analysis for the Rod Ejection Accident (REA) with the assumption that a portion of the fuel fails. The previous REA analysis did not assume fuel failure and, therefore, did not include a dose consequence analysis. The amendment further authorizes the inclusion of related provisions for gas gap release fractions for high-burnup fuel rods (i.e., greater than 54 gigawatt days per metric ton uranium) that exceed the 6.3 kilowatt per foot linear heat generation rate limit detailed in Table 3 of Regulatory Guide 1.183.

A copy of the related Safety Evaluation is enclosed. Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

Dennis J. Galvin

Dennis J. Galvin, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-261

Enclosures:
1. Amendment No. 255 to DPR-23
2. Safety Evaluation

cc: Listserv

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

DUKE ENERGY PROGRESS, LLC

DOCKET NO. 50-261

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 255
Renewed License No. DPR-23

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Duke Energy Progress, LLC (the licensee) (previously Duke Energy Progress, Inc.), dated September 14, 2016, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, by Amendment No. 255, the license is amended to authorize revision to the Updated Final Safety Analysis Report (UFSAR), as set forth in the application dated September 14, 2016. The licensee shall update the UFSAR to incorporate the changes as described in the licensee's application dated September 14, 2016, and the NRC staff's safety evaluation attached to this amendment, and shall submit the revised description authorized by this amendment with the next update of the UFSAR.

3. This license amendment is effective as of its date of issuance and shall be implemented within 120 days from the date of issuance. The UFSAR changes shall be implemented in the next periodic update to the UFSAR in accordance with 10 CFR 50.71(e).

FOR THE NUCLEAR REGULATORY COMMISSION

Undine Shoop, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Date of Issuance: September 29, 2017

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 255 TO

RENEWED FACILITY OPERATING LICENSE NO. DPR-23

DUKE ENERGY PROGRESS, LLC

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

DOCKET NO. 50-261

1.0 INTRODUCTION

By letter dated September 14, 2016 (Reference 1), Duke Energy Progress LLC (previously Duke Energy Progress, Inc.) (Duke Energy or the licensee), submitted a license amendment request (LAR) to modify the alternative source term (AST) methodology at H. B. Robinson Steam Electric Plant Unit No. 2 (HBRSEP). The revision is needed to support the transition from an 18-month to a 24-month fuel cycle. The LAR also includes a dose consequence analysis for the Rod Ejection Accident (REA) to support the assumption that a portion of the fuel fails. The previous REA analysis did not assume fuel failure and therefore did not include a dose consequence analysis. The LAR also includes provisions for gas gap release fractions for high-burnup fuel rods (i.e., greater than 54 gigawatt days per metric ton uranium (GWD/MTU)) that exceed the 6.3 kilowatt per foot (kW/ft) linear heat generation rate limit detailed in Table 3 of Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," dated July 2000 (Reference 2). The LAR identifies changes to the Technical Specification (TS) Bases, the Updated Final Safety Analysis Report (UFSAR), and the Technical Requirements Manual (TRM). By letter dated April 3, 2017, the licensee separately submitted a LAR requesting TS changes for the extended fuel cycle (Reference 3).

2.0 REGULATORY EVALUATION

AST refers to a fission product release from the reactor core that is estimated using more physically based assumptions regarding the composition of the material and the mechanisms for its release during design-basis accidents (DBAs). This is an alternative to the intentionally conservative set of assumptions that most nuclear plants were licensed under, which assumes for example an immediate release. Section 2.3 of this SE provides background on AST requirements. A REA is the postulated failure of the control rod housing mechanism resulting in (1) the ejection of a control rod to the fully withdrawn position and (2) a relatively high rate of reactivity insertion. The REA is analyzed to assure that its effects can neither result in damage to the reactor coolant pressure boundary greater than limited local yielding nor sufficiently

disturb the core, its support structures, or other reactor pressure vessel internals to impair significantly the capability to cool the core.

In performing its technical and safety review, the U.S. Nuclear Regulatory Commission (NRC) staff evaluated the licensee's DBA AST accident analysis for compliance with the following regulations and adherence to the following NRC acceptable dose consequence analysis assumptions and methods as described in the applicable regulatory codes, guides, standards, and approved precedents.

2.1 Regulatory Requirements

Title 10 of the *Code of Federal Regulations* (10 CFR) 50.67 "Accident source term," establishes acceptance criteria for design basis accident radiological analyses. This regulation states, in part, that the applicant's analysis must demonstrate with reasonable assurance that:

(i) An individual located at any point on the boundary of the exclusion area for any two hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 0.25 sievert (Sv) (25 roentgen equivalent man (rem)) total effective dose equivalent (TEDE),

(ii) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release during the entire period of its passage, would not receive a total radiation dose in excess 0.25 Sv (25 rem) TEDE; and

(iii) Adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) TEDE for the duration of the accident.

2.2 Approved Guidance

RG 1.183 provides guidance to licensees of operating power reactors on acceptable applications of ASTs; the scope, nature, and documentation of associated analyses and evaluations; consideration of impacts on analyzed risk; and content of submittals. This guide establishes an acceptable AST and identifies the significant attributes of other ASTs that may be found acceptable by the NRC staff. This guide also identifies acceptable radiological analysis assumptions for use in conjunction with the accepted AST.

Table 3, Footnote 11 of RG 1.183 states, "The release fractions listed here have been determined to be acceptable for use with currently approved LWR [light water reactor] fuel with a peak burnup up to 62,000 MWD/MTU [megawatt days per metric ton of uranium] 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, 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 [boiling water reactor] rod drop accident and PWR [pressurized water reactor] rod ejection accident, the gap fractions are assumed to be 10% for iodines and noble gases."

NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (SRP) Section 6.5.2, "Containment Spray as a Fission Product

Cleanup System" (Reference 4) provides guidance on the fission product removal effectiveness that may be credited for a containment atmosphere fission product cleanup function or system. SRP Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms" (Reference 5), provides DBA-specific TEDE dose criteria.

2.3 Background on Alternative Source Term Requirements

The evaluation of the release of fission products into containment (called "source term") is used for judging the acceptability of both the plant site and the effectiveness of engineered safety features. In the past, power reactor licensees have typically used U. S. Atomic Energy Commission Technical Information Document (TID) 14844, "Calculation of Distance Factors for Power and Test Reactor Sites" (Reference 6), dated March 23, 1962, as the basis for DBA source terms. DBAs are based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events that would result in potential hazards not exceeded by those from any accident considered credible. The DBA offsite radiological dose consequences are evaluated against the guideline dose values, in terms of whole body and thyroid dose, given in 10 CFR 100.11, which references TID-14844.

In December 1999, the NRC issued a new regulation, 10 CFR 50.67, and amended other regulations to allow holders of operating licenses to replace the traditional accident source term methodology used in the DBA analysis with an AST. Section 50.67 of 10 CFR requires a licensee seeking to use an AST to apply for a license amendment and include in the application an evaluation of the consequences of DBAs. When using the provisions of 10 CFR 50.67, the fission product release is assumed to occur over 2 hours as opposed to the TID source term that assumed the release of the entire source term occurs instantaneously. In addition, in the AST, 95% of the radioiodine is assumed to be released as an aerosol, with the remaining 5% as a combination of inorganic and organic vapors. This is in contrast to the original TID source term that prescribed the opposite ratio, 95% of the iodine as vapor and 5% as aerosol.

Guidance for the implementation of the AST methodology is provided in RG 1.183, as discussed above. As part of the implementation of the AST, the TEDE acceptance criterion of 10 CFR 50.67(b)(2) replaces the previous whole body and thyroid dose guidelines of 10 CFR 100.11.

The NRC approved AST implementation for HBRSEP through a series of amendments. By letter dated October 4, 2002 (Reference 7), the NRC issued HBRSEP Amendment No. 195 and revised the TS for HBRSEP to permit a selective implementation of the AST and modify the TS requirement for movement of irradiated fuel and performing core alterations. This TS revision incorporated a reanalysis of the limiting Fuel Handling Accident (FHA) using the AST in accordance with the guidance in RG 1.183.

By letter dated September 24, 2004 (Reference 8), the NRC issued HBRSEP Amendment No. 201 and approved implementation of the AST at HBRSEP for the Main Steam Line Break (MSLB), Steam Generator Tube Rupture (SGTR), Locked Rotor, and the Rod Control Cluster Assembly (RCCA) Withdrawal accidents. The amendment also made revisions to the TS associated with implementation of the AST. The NRC staff took exception to the assumptions used to calculate the containment spray removal constants used in the loss-of-coolant accident (LOCA) dose consequence analysis and therefore denied the LAR with respect to the LOCA dose consequence analysis.

By letter dated July 11, 2006 (Reference 9), the NRC issued HBRSEP Amendment No. 207 and approved the implementation of the AST methodology for the LOCA at HBRSEP. There were

no associated changes to the TS. The licensee made changes to the assumptions used to calculate the containment spray removal constants that addressed the NRC concerns related to HBRSEP Amendment No. 201.

By letter dated April 29, 2014 (Reference 10), the licensee provided a description of changes that were implemented pursuant to 10 CFR 50.59, "Changes, tests and experiments," at HBRSEP between April 1, 2012, and April 1, 2014. The April 29, 2014 submittal documents that based on CR inleakage testing in 2003 and 2012, the licensee determined that the actual inleakage with HVS-1 running is less than the assumed inleakage with HVS-1 secured in the CR dose analyses. Therefore, a manual action to reduce inleakage into the control room (CR) assumed in several CR dose analyses (i.e., securing the Auxiliary Building Supply Fan HVS-1 within 1 hour of the receipt of a Safety Injection Signal or following an FHA) is no longer necessary to support the associated CR dose analyses. Thus, using a higher inleakage for the first hour of a CR dose analysis is conservative relative to measured values. The licensee used the 10 CFR 50.59 process to remove the manual action from an operations procedure but retained the 1-hour delay for reduced inleakage in the existing CR dose analyses. The REA analysis included in the LAR does not include the 1-hour delay in reduced inleakage. The applicability of the April 29, 2014 submittal to specific CR dose analyses is discussed below.

3.0 TECHNICAL EVALUATION

3.1 Radiological Consequences of Design Basis Accidents

As described in the regulatory evaluation section above, the licensee's UFSAR dose consequence analyses are based on the AST following the guidance provided in RG 1.183. The following technical evaluation will focus on the changes requested in the LAR (Reference 1). This safety evaluation (SE) will also include excerpts from previous SEs associated with AST license amendments in order to provide a complete assessment of each of the revised design basis dose consequence analyses incorporating the AST.

The LAR identified three proposed changes to the dose consequence analyses for the DBAs described in the UFSAR.

1. The licensee proposed changes in the source term used in the dose consequence analyses as a result of the transition from an 18-month cycle length to a 24-month cycle length.

2. The licensee proposed changes in the assumed gap release fractions for fuel rods with burnup greater than 54 GWD/MTU that exceed the 6.3 kW/ft linear heat generation rate (LHGR). To accommodate these high burnup fuel rods, the licensee has proposed gap release fractions higher than those detailed in Table 3 of RG 1.183.

3. In addition, the licensee provided a new DBA dose consequence analysis for the REA assuming that a degree of fuel damage occurs. Previous assessments of the REA did not assume fuel damage and, therefore, did not include a dose consequence analysis.

In order to support the licensing changes discussed in the LAR, the licensee analyzed the following accidents employing the AST as described in RG 1.183.

1. Loss of Coolant Accident (LOCA)

2. Fuel Handling Accident (FHA)

3. Main Steam Line Break (MSLB) Accident

4. Locked Rotor Accident (LRA)

5. Single Rod Withdrawal (SRW) Accident

6. Rod Ejection Accident (REA)

The SGTR accident evaluated in UFSAR Section 15.6.3, "Steam Generator Tube Rupture (SGTR)," was not impacted by the proposed changes described in this LAR since the SGTR accident does not predict fuel damage. Since the licensee does not predict SGTR fuel failure for a 24 month fuel cycle, the SGTR dose consequence analysis is not affected by this LAR. The SGTR analysis is based on the limiting reactor coolant system (RCS) activity as governed by TSs with the associated iodine spiking described in RG 1.183.

The licensee evaluated each DBA integrated TEDE dose at the Exclusion Area Boundary (EAB) for the worst 2-hour period following the onset of the accident. The integrated TEDE doses at the outer boundary of the low population zone (LPZ) and the integrated dose to a CR operator were evaluated for the 30-day duration of the accident evaluation period. The dose consequence analyses were performed by the licensee using the "RADTRAD: Simplified Model for RADionuclide Transport and Removal And Dose Estimation, Version 3.03, computer code." NRC sponsored the development of the RADTRAD radiological consequence computer code, as described in NUREG/CR-6604 (Reference 11). The RADTRAD code was developed by Sandia National Laboratories for the NRC. The code estimates transport and removal of radionuclides and radiological consequence doses at selected receptors. The NRC staff performed independent sensitivity dose evaluations using the RADTRAD computer code. The results of the evaluations performed by the licensee, as well as the applicable dose acceptance criteria from RG 1.183, are shown in Table 1 of the Attachment to this SE. The CR atmospheric dispersion factors, offsite atmospheric dispersion factors, CR data and assumptions used in the licensee's evaluations are included in Tables 2, 3, and 4 of the Attachment to this SE, respectively.

RG 1.183, Regulatory Position 3.1, "Fission Product Inventory," states that, "The inventory of fission products in the reactor core and available for release to the containment should be based on the maximum full power operation of the core with, as a minimum, current licensed values for fuel enrichment, fuel burnup, and an assumed core power equal to the current licensed rated thermal power times the ECCS [emergency core cooling system] evaluation uncertainty. The period of irradiation should be of sufficient duration to allow the activity of dose-significant radionuclides to reach equilibrium or to reach maximum values. The core inventory should be determined using an appropriate isotope generation and depletion computer code such as ORIGEN 2 or ORIGEN-ARP."

In accordance with RG 1.183, the licensee generated the core and worst case fuel assembly radionuclide inventories for use in determining source term inventories using ORIGEN-S and ORIGEN-ARP. In order to cover the transition from 18-month cycles to 24-month cycles, the licensee developed a conservative source term based on results from both the 18-month fuel cycle source term and the 24-month fuel cycle source term. The licensee accomplished this by taking the maximum activity for either the 18-month or the 24-month cycles for each individual isotope. The licensee provided a table of the HBRSEP source term, including the 60 isotopes

required by NUREG/CR-6604. The licensee multiplied the source term by 1.003 to reflect operation at 100.3% of current rated power (100.3% of 2339 Megawatts thermal (MW_t), or 2346 MW_t), which accounts for the uncertainty in measured core power. The NRC staff considers the licensee's method to be a conservative approach and finds that it is acceptable for the purpose of determining a bounding source term for the evaluation of DBAs.

The licensee used committed effective dose equivalent and effective dose equivalent dose conversion factors (DCFs) from Federal Guidance Reports (FGRs) 11 and 12 (References 12 and 13, respectively) to determine the TEDE dose as is required for AST evaluations. The use of ORIGEN and DCFs from FGR-11 and FGR-12 follows the guidance in RG 1.183 and is, therefore, acceptable to the NRC staff.

3.1.1 Fission Gas Gap Fractions

3.1.1.1 Fission Gas Gap Fractions Background

For design basis non-LOCA events in which fuel damage is assumed or predicted to occur, the quantity of each isotope released from the fuel is an important input to the dose calculation. To determine the quantity of each isotope released, gap fractions and the number of failed fuel rods are used to determine the accident source term. Gap fractions describe, within a fuel rod, the fraction of the total inventory of an isotope that resides in the void region of the fuel rod (i.e., gap between the fuel pellet and the fuel cladding and plenum). When fuel damage is assumed to occur, the gas in the gap, containing these isotopes, is assumed to be released from the fuel for the purpose of modeling and predicting dose.

RG 1.183 includes a table of acceptable gap fractions, repeated here:

Table 3 from RG 1.183:
Non-LOCA Fraction of Fission Product Inventory in Gap

Group	Fraction
I-131	0.08
Kr-85	0.10
Other Noble Gases	0.05
Other Halogens	0.05
Alkali Metals	0.12

This table also references Footnote 11, which describes the range of burnups and LHGRs where this table applies. This footnote reads:

> 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, 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.

The LAR seeks approval of a method that may be used to determine fission gas gap fractions for fuel rods that exceed the burnup and LHGR limits listed in RG 1.183, Table 3, Footnote 11.

3.1.1.2 Fission Gas Gap Fractions Precedent

The NRC staff previously reviewed and approved a similar amendment regarding fission gas gap fractions for McGuire Nuclear Stations Unit Nos. 1 and 2, Catawba Nuclear Station Unit Nos. 1 and 2, and Oconee Nuclear Station Unit Nos. 1, 2, and 3 (LAR: Reference 14: Safety Evaluation: Reference 15). This previous review (Reference 15) included an audit (Reference 16) of the licensee's methods for evaluating the fission gas gap fraction for rods that exceeded the limits in Footnote 11. The previous review found the licensee's fission gas gap fractions acceptable.

During the audit, NRC staff performed confirmatory calculations using FRAPCON 4.0 (Reference 17) to provide assurance that Duke Energy's proposed gap fraction multipliers were conservative and appropriate, and found them to be so. NRC staff also examined the licensee's use of the COPERNIC code (Reference 18) and the gapfrac macro to ensure the underlying engineering calculations were consistent with the American Nuclear Society (ANS) standard ANS 5.4. These were found to be acceptable.

3.1.1.3 Fission Gas Gap Fractions Staff Evaluation

In this LAR (Reference 1), the licensee is proposing new gap fractions for AST determination in rods that exceed the 6.3 kW/ft peak rod average power after achieving burnups over 54 GWD/MTU. The licensee has calculated new gap fractions to support these higher LHGRs using approved codes and the ANS 5.4 standard as updated in 2011 (Reference 19).

The NRC staff approved the use of this approach in a previous SE (Reference 15) for Duke Energy, as described in the previous section. The HBRSEP LAR makes a similar request with two changes. First, the previously approved LAR (Reference 14) used both the ANS 5.4 (1982) and the updated ANS 5.4 (2011) standards for calculating fission gas gap fractions, and used the most conservative results for each isotope group. The HBRSEP LAR only uses the 2011 standard. The NRC staff finds this acceptable, as the approach for calculating fission gas gap fractions in the ANS 5.4 (2011) standard is consistent with the approach proposed in a draft revision to RG 1.183, DG-1199 (Reference 20).

The other notable change is the use of a different LHGR versus burnup limit. The NRC staff finds this limit to be acceptable since the limit remains within the scope of the previous review, as described below.

The licensee calculated the gap inventory of the various isotopes by plugging nodal fuel temperatures from COPERNIC into a Visual Basic for Applications macro named gapfrac. Gapfrac applies the ANS 5.4 (2011) standard, and was reviewed during the previous audit (Reference 16). The COPERNIC calculations take into account the burnup and LHGR rates described by the licensee. The resulting isotope gap fractions are the same or lower than RG 1.183, Table 3, with the exception of Kr-85, Cs-134, and Cs-137. For these isotopes, the ratio between the gap fractions calculated by the licensee and those in RG 1.183, Table 3 is shown in the table below, taken from the LAR, Enclosure I, Table 6. The bounding gap fraction is taken from the LAR, Enclosure I, Table 7.

Isotope	RG 1.183 Table 3 Value	Robinson HTP* Fuel Calculated Maximum Gap Fraction	Ratio	Bounding Gap Fraction
Kr-85	0.10	0.168	1.68	0.30
Cs-134	0.12	0.238	1.98	0.36
Cs-137	0.12	0.238	1.98	0.36

*High Thermal Performance

In order to provide an added degree of conservatism, the licensee applied a multiplier of 3 to the RG 1.183 values for all three of these isotopes, as shown in the final column. The values for the remaining isotopes remained consistent with the values in RG 1.183, Table 3.

As all of the methods used in the calculation of the fission gas gap fractions have been previously reviewed and found acceptable, and the method used in this LAR is within the applicable scope of those reviews, for fuel rods that exceed the burnup and LHGR limits listed in RG 1.183, Table 3, Footnote 11, the NRC staff finds the fission gas gap fractions acceptable for use within the LHGR and burnup limits described in the LAR, Enclosure I, Table 3.

3.2.1 Loss-of-Coolant Accident

The radiological consequence design basis LOCA analysis is a deterministic evaluation based on the assumption of a major rupture of the primary RCS piping. The accident scenario assumes the deterministic failure of the ECCS to provide adequate core cooling that results in a significant amount of core damage as specified in RG 1.183. This general scenario does not represent any specific accident sequence, but is representative of a class of severe damage incidents that were evaluated in the development of the RG 1.183 source term characteristics. Such a scenario would be expected to require multiple failures of systems and equipment and lies beyond the severity of incidents evaluated for design-basis transient analyses.

A LOCA evaluation using the AST is a complete and instantaneous circumferential severance of the primary RCS piping, which would result in the maximum fuel temperature and primary containment pressure among the full range of LOCAs. Due to the postulated loss of core cooling, the fuel heats up, resulting in the release of fission products. The fission product release is assumed to occur in phases over a 2-hour period.

When using the AST for the evaluation of a design-basis LOCA for a PWR, it is assumed that the initial fission product release to the containment will last for 30 seconds and will consist of the radioactive materials dissolved or suspended in the RCS liquid. After 30 seconds, fuel damage is assumed to begin and is characterized by clad damage that releases the fission product inventory assumed to reside in the fuel gap. The fuel gap release phase is assumed to continue until 30 minutes after the initial breach of the RCS. As core damage continues, the gap release phase ends and the early in-vessel release phase begins. The early in-vessel release phase continues for the next 1.3 hours. The licensee used the LOCA source term release fractions, timing characteristics, and radionuclide grouping as specified in RG 1.183 for evaluation of the AST.

3.2.1.1 Source term

The HBRSEP LOCA analysis assumes that iodine will be removed from the containment atmosphere by both containment sprays and natural diffusion to the containment walls. As a

result of these removal mechanisms a large fraction of the released activity will be deposited in the containment sump. The sump water will retain soluble gaseous and soluble fission products such as iodine and cesium, but not noble gases. The guidance from RG 1.183 specifies that the iodine deposited in the sump water can be assumed to remain in solution as long as the containment sump pH is maintained at or above 7.

As documented in the UFSAR Section 15.6.5, "Loss-of-Coolant Accidents," the licensee's LOCA analysis provides sump pH control through plant procedures that provide guidance to maintain the sump pH above 8.5 through the use of Sodium Hydroxide from the Spray Additive Tank.

3.2.1.2 Assumptions on transport in the primary containment

3.2.1.2.1 Containment mixing, natural deposition and leak rate

In accordance with RG 1.183, the licensee assumed that the activity released from the fuel is mixed instantaneously and homogeneously throughout the free air volume of the containment. The licensee used the core release fractions and timing as specified in RG 1.183, Table 2 with the termination of the release into containment set at the end of the early in-vessel phase.

To ensure adequate mixing between the sprayed and unsprayed regions, the containment is equipped with two safety related containment cooling fans each rated at 65,000 cubic feet per minute (cfm). Consistent with UFSAR Section 15.6.5, the licensee credited the operation of these fans beginning at 76 seconds post-LOCA.

Consistent with UFSAR Section 15.6.5, the licensee credited the reduction of airborne radioactivity in the containment by natural deposition. The licensee credited a particulate iodine natural deposition removal coefficient of 0.1 hr^{-1} for the unsprayed containment regions and for the sprayed regions during periods when the sprays are not in operation. The licensee did not credit the removal of elemental iodine or organic iodide by natural deposition.

In the LAR (Reference 1), the licensee chose not to continue to take credit for diffusiophoresis, currently in UFSAR Section 15.6.5, as approved in HBRSEP Amendment No. 207, issued July 11, 2006 (Reference 9). Diffusiophoresis acts to remove airborne particulates by: a) the condensation of containment steam produced by the large break LOCA onto airborne particulates, b) the entrainment and removal of particulates via steam condensation onto passive heat sinks; or c) condensation and removal of particulates by active, forced heat removal mechanisms, such as fan coolers, or the condensation of steam caused by the cooling action of the sprays. Since taking credit for diffusiophoresis would reduce the airborne particulate activity in the containment atmosphere and, hence, lower the dose consequence from containment leakage, the licensee's decision not to credit this phenomena is conservative and, therefore, acceptable to the NRC staff.

RG 1.183, Regulatory Position 3.7 states that, "The primary containment should be assumed to leak at the peak pressure technical specification leak rate for the first 24 hours. For PWRs, the leak rate may be reduced after the first 24 hours to 50% of the technical specification leak rate." Accordingly, the licensee retained the UFSAR Section 15.6.5 assumption of a containment leak rate of 0.1% weight per day for the first 24 hours, after which the containment leak rate is reduced to 0.05% weight per day for the duration of the accident.

3.2.1.2.2 Containment spray assumptions

RG 1.183, Appendix A, "Assumptions for Evaluating the Radiological Consequences of a LWR Loss-of-Coolant Accident," Regulatory Position 3.3 states that, "The containment building atmosphere may be considered a single, well-mixed volume if the spray covers at least 90% of the volume and if adequate mixing of unsprayed compartments can be shown." In addition, SRP Section 6.5.2, Subsection III.1.c (Reference 4) states in part, "The containment building atmosphere may be considered a single, well-mixed space if the spray covers regions comprising at least 90% of the containment building space and if a ventilation system is available for adequate mixing of any unsprayed compartments."

The HBRSEP containment volume is 1,958,526 cubic feet (ft^3), with a sprayed region volume of 1,623,618.1 ft^3 for train A and a sprayed region volume of 1,596,198.7 ft^3 for train B. For either train, the sprayed region represents approximately 80% of the total containment volume. Therefore, consistent with the UFSAR Section 15.6.5, the licensee used a two volume model to represent the sprayed and unsprayed regions of the containment. Since the spray coverage is slightly less for train B, for conservatism the licensee modeled the spray removal using train B coverage.

Using the guidance from SRP Section 6.5.2, the licensee modeled the removal of iodine in both aerosol (also referred to as particulate) and elemental (also referred to as vapor) forms. Regarding aerosol iodine, the licensee determined that the aerosol iodine removal rate from the effects of the containment spay system, which actuates 3 minutes after the LOCA, is 3.427 per hour until a decontamination factor (DF) of 50 is reached at 2.76 hours post-LOCA. The train B aerosol iodine removal rate of 3.427 per hour represents the UFSAR Section 15.6.5 removal rate of 3.627 per hour minus the 0.2 per hour removal rate from the UFSAR Section 15.6.5 credit due to diffusiophoresis. As previously mentioned, the licensee has chosen not to use the UFSAR Section 15.6.5 credit due to diffusiophoresis in the LAR. Therefore, the licensee's use of an aerosol iodine removal rate of 3.427 per hour is consistent with the UFSAR Section 15.6.5 credit for spray aerosol iodine removal (excluding diffusiophoresis).

Consistent with UFSAR Section 15.6.5, the licensee modelled a ten minute interval from 77 minutes until 87 minutes post-LOCA during which the containment spray is stopped in order to accommodate the switchover to the recirculation mode of operation. During this 10-minute interval, the licensee modeled the aerosol iodine removal rate based on a natural deposition rate of 0.1 per hour. In accordance with guidance in SRP Section 6.5.2, after the aerosol iodine spray removal DF of 50 is reached at 2.76 hours, the licensee reduced the aerosol iodine spray removal coefficient by a factor of 10 to 0.3427 per hour until the spray is terminated at 2.78 hours post-LOCA.

Regarding elemental iodine, in accordance with the guidance in SRP Section 6.5.2, the licensee limited the removal rate constant for elemental iodine to 20 per hour during the period of spray operation. The licensee did not credit elemental iodine removal prior to spray initiation, during the 10-minute period for transfer to recirculation (i.e., from 77 minutes until 87 minutes post-LOCA) or after the maximum DF of 200 is reached at 2.04 hours post-LOCA.

The NRC staff reviewed the credits the licensee did and did not apply for aerosol and elemental iodine removal from the operation of the containment spay system and found that the analysis followed the applicable regulatory guidance from SRP Section 6.5.2, is conservative and is therefore acceptable.

3.2.1.3 Assumptions on Engineered Safety Feature System Leakage

Engineered Safety Feature (ESF) leakage develops when ESF systems circulate containment sump water outside containment and leaks develop through packing glands, pump shaft seals and flanged connections. To evaluate the radiological consequences of ESF leakage, the licensee used the deterministic approach as prescribed in RG 1.183. This approach assumes that, except for the noble gases, all of the fission products released from the fuel mix instantaneously and homogeneously in the containment sump water. Except for iodine, all of the radioactive materials in the containment sump are assumed to be in aerosol form and are retained in the liquid phase. As a result, the licensee assumed that the fission product inventory available for release from ESF leakage consists of 40% of the core inventory of iodine. This amount is the combination of 5% released to the containment sump water during the gap release phase and 35% released to the containment sump water during the early in-vessel release phase. This source term assumption is conservative in that 100% of the radioiodines released from the fuel are assumed to reside in both the containment atmosphere and in the containment sump concurrently.

For the LOCA analysis, the licensee used a value of 2 gallons per hour of ESF leakage, representing two times the maximum permitted in the TRM. The use of an ESF leakage value two times the site specific leakage limit is consistent with the guidance specified in RG 1.183, Appendix A, Item 5.2. As stated above, actual ESF leakage would not begin until after the recirculation phase of the accident begins. The licensee assumed that ESF leakage will start at 21 minutes post-LOCA and continue for the 30-day duration of the accident evaluation. The NRC staff notes that the licensee did not change any of the UFSAR Section 15.6.5 assumptions regarding the dose consequence analysis of ESF leakage for this LAR.

3.2.1.3.1 Assumptions on Engineered Safety Feature System Back-Leakage to the Refueling Water Storage Tank

Due to the location of the Refueling Water Storage Tank (RWST) with respect to the CR, for conservative reasons, the licensee maintained the UFSAR Section 15.6.5 assumption that ESF system leakage is from the residual heat removal (RHR) heat exchanger room (located in the Auxiliary Building), which has a greater CR atmospheric dispersion factor. The NRC staff notes that addressing a dose consequence analysis for ESF system back-leakage to the RWST is important in those cases where credit is taken for filtration of the normal ESF leakage and the RWST back-leakage is released from the RWST with no filtration or when the RWST is in close proximity to the CR. The licensee did not credit filtration of the normal ESF leakage. In addition, due to the location of the RWST relative to the CR, releases from the RWST will have a more favorable atmospheric dispersion factor than releases from the RHR heat exchanger room. Therefore, the licensee's assumption that all the ESF leakage releases are from the RHR heat exchanger is conservative.

3.2.1.4 Control Room Ventilation Assumptions

The NRC staff notes that the licensee did not change any of the UFSAR Section 15.6.5 CR ventilation assumptions regarding the LOCA dose consequence analysis for this LAR. Consistent with UFSAR Section 15.6.5, the licensee assumed that the normal ventilation system would be operating for 35 seconds post-LOCA before a safety injection signal would result in the automatic initiation of the CR's emergency ventilation system. During this period of normal operation, in addition to the 400 cfm of normal unfiltered intake flow, the licensee assumed an unfiltered inleakage rate into the CR envelope of 170 cfm. After the safety injection signal

initiates the CR emergency ventilation system the intake flow of 400 cfm is directed to the CR emergency filtration system, which is comprised of both particulate filters and a charcoal adsorber. In addition, the CR emergency filtration system recirculates CR air at a rate of 2,600 cfm.

In accordance with RG 1.183, the licensee analyzed the LOCA dose consequences assuming a loss of offsite power (LOOP). With offsite power available, the Hagan Room,[1] which is adjacent to the CR, remains at a negative pressure relative to the CR due to the operation of the auxiliary building exhaust fan HVE-7. The UFSAR Section 15.6.5 CR dose consequence analysis assumes that, as a result of the LOOP and the loss of HVE-7, for the first hour the Hagan Room would be at a higher pressure than the CR resulting in an increased CR unfiltered inleakage of 170 cfm. The Hagan Room is assumed to remain in this condition until operators can take manual actions that will result in the reduction of the pressure in the Hagan Room to below that of the CR. The licensee's analysis assumed that it would take approximately 1 hour to implement the manual actions to reduce the pressure in the Hagan Room. Therefore, the CR LOCA dose consequence analysis assumes that for the first hour of the accident the CR unfiltered inleakage rate is 170 cfm. After 1 hour, the unfiltered inleakage is assumed to be reduced by 70 cfm to 100 cfm when the inleakage from the Hagan Room is reduced.

The NRC staff notes that the letter dated April 29, 2014 (Reference 10), documents, as discussed above, that based on CR inleakage testing in 2003 and 2012, the licensee determined that the actual inleakage to the CR if no manual action is taken (and the associated 1-hour delay) is less than the assumed inleakage to CR after 1 hour. Thus, using a higher inleakage for the first hour of a CR dose analysis is conservative relative to measured values. Therefore maintaining the assumption of a 1-hour delay before the reduction in CR unfiltered inleakage from 170 cfm to 100 cfm is conservative for the LOCA analysis of CR habitability.

3.2.1.5 Loss-of-Coolant Accident Conclusions

The licensee evaluated the radiological consequences resulting from the postulated LOCA and concluded that the radiological consequences at the EAB, LPZ, and CR are within the dose acceptance criteria specified in 10 CFR 50.67 as well as the accident specific dose criteria specified in SRP Section 15.0.1. The NRC staff's review has found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE. The assumptions that the NRC staff found acceptable are in Table 5 of the Attachment to this SE and the licensee's calculated dose results are in Table 1 of the Attachment to this SE. The NRC staff performed independent sensitivity dose evaluations as necessary to ensure a thorough understanding of the licensee's methods. The NRC staff finds that the EAB, LPZ, and CR doses estimated by the licensee for the LOCA meet the applicable accident dose acceptance criteria and are, therefore, acceptable.

3.2.2 Fuel-Handling Accident

The licensee's FHA analysis involved the utilization of the AST and an assessment of two cases. The first case assumes an FHA occurring within containment. The second case assumed an FHA occurring within the fuel-handling building (FHB). In both cases, the licensee

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[1] The Hagan Room is a room adjacent to the CR and contains analog instrumentation, controls, associated test panels, and some electrical equipment that supports the CR lighting. The room is referred to as the Hagan Room by the licensee based on the brand name of the analog instrumentation system in the room.

assumed that dropping a fuel assembly results in damage to all of the fuel rods in the dropped assembly.

3.2.2.1 Source Term

The fission product inventory that constitutes the source term for this event is the gap activity in the fuel rods assumed to be damaged as a result of the postulated design basis FHA. Volatile constituents of the core fission product inventory migrate from the fuel pellets to the gap between the pellets and the fuel rod cladding during normal power operations. The fission product inventory in the fuel rod gap of the damaged fuel rods is assumed to be instantaneously released to the surrounding water as a result of the accident. In the LAR, the licensee increased the minimum decay time allowed prior to fuel movement from the current UFSAR Section 15.7.4, "Design Basis Fuel Handling Accidents," value of 56 hours to the proposed value of 116 hours. This change reduces operational flexibility but allows for more decay time thereby reducing the potential FHA source term.

Fission products released from the damaged fuel are decontaminated by passage through the overlaying water in the reactor cavity or spent fuel pool (SFP) depending on their physical and chemical form. Following the guidance in RG 1.183, Appendix B, "Assumptions for Evaluating the Radiological Consequences of a Fuel Handling Accident," Regulatory Position 1.3, the licensee assumed that; the chemical form of radioiodine released from the fuel to the SFP consists of 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide, the CsI released from the fuel is assumed to completely dissociate in the pool water, and because of the low pH of the pool water, the CsI re-evolves as elemental iodine. This results in a final iodine distribution of 99.85% elemental iodine and 0.15% organic iodine. The licensee assumed that the release to the pool water and the chemical redistribution of the iodine species occurs instantaneously. The licensee assumed the gap inventory of the damaged fuel rods was in an assembly that had been operated at 1.8 times core average power (radial peaking factor of 1.8).

As corrected by item 8 of Regulatory Issue Summary (RIS) 2006-04 (Reference 21), RG 1.183, Appendix B, Regulatory Position 2, should read, "If the depth of water above the damaged fuel is 23 feet or greater, the decontamination factors for the elemental and organic species are 285 and 1, respectively, giving an overall effective decontamination factor of 200 (i.e., 99.5% of the total iodine released from the damaged rods is retained by the water)." This difference in decontamination factors for elemental (99.85%) and organic iodine (0.15%) species results in the iodine above the water being composed of 70% elemental and 30% organic species.

In accordance with RG 1.183, Appendix B, Regulatory Position 2, the licensee credited an overall iodine DF of 200 for a water cover depth of 23 feet for the FHA inside containment. For the FHA in the fuel building, the licensee credited a DF of 138 due to the reduced water cover of 21 feet. The NRC staff confirmed the acceptability of the use of a DF of 138 for a water cover of 21 feet using the referenced methodology in RG 1.183, Appendix B, Regulatory Position 2. In accordance with RG 1.183, the licensee assumed that particulate radionuclides will be retained by the water in the fuel pool or reactor cavity (i.e., infinite DF). In accordance with RG 1.183, the licensee did not credit decontamination from water scrubbing for the noble gas constituents of the gap activity.

As stated in RG 1.183, the release fractions associated with the LWR core inventory released into containment for the DBA LOCA and non-LOCA events have been determined to be acceptable for use with currently approved LWR fuel with a peak burnup of 62,000 MWD/MTU

provided that the maximum LHGR does not exceed 6.3 kW/ft peak rod average power for burnups exceeding 54,000 MWD/MTU.

The licensee evaluated the FHA assuming that up to 35 rods per assembly exceed the maximum LHGR of 6.3 kW/ft, detailed in RG 1.183, Table 3, but remain below a new maximum LHGR of 7.0 kW/ft. In order to accommodate this increase in LHGR, the licensee tripled the release fractions for the long lived isotopes of Kr-85, Cs-134, and Cs-137 from the values in RG 1.183, Table 3. As discussed in Section 3.1.1.3 of this SE, the licensee calculated gap fractions for the projected power history, and showed that the gap fractions will remain below two times the RG 1.183, Table 3 values for Kr-85, Cs-134, and Cs-137. However for additional conservatism, the licensee applied a bounding multiplier of three to the RG 1.183, Table 3 values for these isotopes. For all other isotopes, the licensee used the release fractions outlined in RG 1.183, Table 3. As discussed in Section 3.1.1 of this SE, the NRC staff finds the fission gas gap fractions for fuel rods that exceed the burnup and LHGR limits listed in RG 1.183, Table 3, Footnote 11, acceptable for use within the LHGR and burnup limits as described in the LAR, Enclosure I, Table 3.

3.2.2.2 Transport

For the FHA occurring in the FHB, the licensee assumed that all activity released from the SFP is exhausted as a ground level release to the environment over a 2-hour period through the FHB air handling and filtration system. The licensee credited the FHB filtration system for the removal of 90% of the elemental iodine and 70% of the organic iodide released from the SFP water.

For the FHA occurring in the containment, the licensee assumed that all activity released from the reactor cavity is exhausted as a ground level release to the environment over a 2-hour period. The licensee did not credit the containment air handling system or containment closure for the FHA dose consequence analysis.

3.2.2.3 Control Room Habitability for the Fuel-Handling Accident

The licensee assumed that the CR emergency filtration system will become operational at 1 hour following the onset of the accident. Prior to the initiation of the CR emergency ventilation system, 400 cfm of unfiltered intake air enters the CR through the normal ventilation system. In addition, prior to the initiation of the CR emergency ventilation, the licensee assumed an unfiltered inleakage of 300 cfm. After 1 hour, the CR emergency ventilation system is actuated by manual operator action resulting in 400 cfm of filtered pressurization intake as well as 2600 cfm of filtered recirculation. During the operation of the CR emergency ventilation system, unfiltered inleakage into the CR was assumed to be 230 cfm. The NRC staff notes that the licensee made no changes to the UFSAR Section 15.7.4 assumptions for CR habitability for the FHA analysis associated with this LAR.

3.2.2.4 Fuel-Handling Accident Conclusions

The licensee evaluated the radiological consequences resulting from a postulated FHA and concluded that the radiological consequences at the EAB, LPZ, and CR are within the dose acceptance criteria specified in 10 CFR 50.67 and the accident specific dose criteria specified in SRP Section 15.0.1. The NRC staff finds that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE. In addition as discussed in Section 3.1.1 of this SE, the NRC staff finds the fission gas gap

fractions for fuel rods that exceed the burnup and LHGR limits listed in RG 1.183, Table 3, Footnote 11, acceptable for use within the LHGR and burnup limits as described in the LAR, Enclosure I, Table 3. The assumptions found acceptable to the NRC staff are in Table 6 of the Attachment to this SE and the licensee's calculated dose results are in Table 1 of the Attachment to this SE. The NRC staff performed independent sensitivity dose evaluations as necessary to ensure a thorough understanding of the licensee's methods. The NRC staff finds that the EAB, LPZ, and CR doses estimated by the licensee for the FHA meet the applicable acceptance criteria and are therefore acceptable.

3.2.3 Primary Coolant Pump Locked Rotor Accident

The accident considered begins with the instantaneous seizure of a reactor coolant pump rotor that causes a rapid reduction in the flow through the affected RCS loop. The sudden decrease in core coolant flow causes a reactor trip on a low primary loop flow signal. The low coolant flow causes a degradation of core heat transfer, resulting in localized temperature and pressure changes in the core. The LRA is described in UFSAR Section 15.3.2, "Reactor Coolant Pump Shaft Seizure (Locked Rotor)."

3.2.3.1 Source Term

The licensee assumed that the instantaneous seizure of the reactor coolant pump rotor associated with the LRA results in 12 fuel assemblies experiencing a departure from nucleate boiling (DNB) that results in fuel clad damage. This represents a change in the UFSAR Section 15.3.2 LRA analysis, which assumes that 17 fuel assemblies experience DNB. The licensee maintained the UFSAR Section 15.3.2 assumption that the LRA will not result in any fuel melting. Therefore, for this LAR the source term available for release is the gap activity in the 12 fuel assemblies that experience DNB. The licensee determined that none of the 12 fuel assemblies that experience DNB will exceed the 6.3 kW/ft limitation on the gap activity percentages listed in RG 1.183, Table 3. Therefore, in accordance with RG 1.183, 5% of the core inventory of iodine and noble gas is assumed to be in the fuel-clad gap, with the exception of 8% assumed for I-131 and 10% assumed for Kr-85. The licensee included 12% of the core cesium and rubidium consistent with RG 1.183, Table 3. The licensee applied a radial peaking factor of 1.8 for the assessment of the source term for the LRA.

3.2.3.2 Transport

Activity from the fuel cladding damage is transported to the secondary side due to primary-to-secondary side leakage. The licensee assumed a total primary-to-secondary leak rate equal to 450 gallons per day (gpd). This represents a change in the UFSAR Section 15.3.2 primary-to-secondary side leakage of 432 gpd. This is a conservative change that is discussed in the section on the MSLB accident of this SE. A LOOP is assumed to occur concurrently with the reactor trip. Because of the LOOP, condenser cooling is lost, which results in releases to the environment associated with secondary coolant steaming from the steam generators.

RG 1.183 allows for a reduction in the amount of activity released to the environment based on partitioning of nuclides between the liquid and gas states of water for this release path. In accordance with RG 1.183, the licensee included a partition coefficient of 100 for halogens and alkali metal releases from the steam generators. Because of their volatility, 100% of the noble gases are assumed to be released without holdup, reduction or mitigation. Consistent with UFSAR Section 15.3.2, the licensee assumed that the steaming release from either the steam generator atmospheric relief valves or safety valves ends after 53.2 hours, which is the time

required for one train of the RHR System to establish adequate shutdown cooling to terminate releases from the steam generators.

The licensee's analysis conforms to RG 1.183, Appendix G, "Assumptions for Evaluating the Radiological Consequences of a PWR Locked Rotor Accident." The licensee assumed the same CR ventilation assumptions as was used for the FHA. The NRC staff notes that the licensee made no changes to the UFSAR Section 15.3.2 assumptions for CR habitability for the LRA analysis associated with this LAR.

3.2.3.3 Locked Rotor Accident Conclusions

The licensee evaluated the radiological consequences resulting from the postulated LRA and concluded that the radiological consequences at the EAB, LPZ, and CR are within the dose acceptance criteria specified in 10 CFR 50.67 and the accident dose acceptance criteria specified in SRP Section 15.0.1. The NRC staff found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE. The assumptions the NRC staff found acceptable are in Table 7 of the Attachment to this SE and the licensee's calculated dose results are in Table 1 of the Attachment to this SE. The NRC staff performed independent sensitivity dose evaluations as necessary to ensure a thorough understanding of the licensee's methods. The EAB, LPZ, and CR doses estimated by the licensee for the LRA meet applicable accident dose acceptance criteria and are, therefore, acceptable.

3.2.4 Single Rod Withdrawal Accident

As stated in the HBRSEP UFSAR Section 15.4.3.1.4, "Radiological Consequences," the single RCCA rod withdrawal event causes an insertion of positive reactivity that results in a power excursion transient and may cause fuel damage. Due to the pressure differential between the primary and secondary systems and assumed steam generator tube leakage, fission products are discharged from the primary into the secondary system. A portion of this radioactivity is released to the outside atmosphere through either the steam generator atmospheric relief valves or safety valves. In addition, radioactivity is contained in the primary and secondary coolant before the accident, and some of this activity is released to the atmosphere as a result of steaming from the steam generators following the accident. The SRW accident is described in UFSAR Section 15.4.3.1, "Withdrawal of a Single Full-Length RCCA."

3.2.4.1 Source term

The licensee performed an assessment of the consequences of a postulated SRW accident. The amount of radioactivity released as a result of this accident was based upon the number of fuel rods experiencing DNB, the number of fuel rods that reach or exceed the initiation temperature for fuel melting, and the radial peaking factor.

The UFSAR Section 15.4.3.1 analysis assumed one fuel assembly experienced DNB thereby releasing its gap activity and that three other assemblies reached or exceeded the initiation temperature for fuel melt. For this LAR, the licensee modified the UFSAR Section 15.4.3.1 assumptions for the degree of fuel damage resulting from an SRW accident. The revised analysis assumes that four assemblies experience DNB, thereby releasing their gap activity but no fuel assemblies reach the initiation temperature for fuel melt.

3.2.4.2 Transport

Radioactivity from reactor coolant would enter secondary coolant as a result of primary-to-secondary leakage. Radioactivity in the secondary-side coolant would be released to the environment based upon the steaming rate and the partition coefficient for the particular nuclide group. Primary-to-secondary leakage was assumed to continue until the primary-side pressure was reduced below that of the secondary side. The release of radioactivity from the steam generators was assumed to be terminated when the RHR system was placed in operation.

RG 1.183, Appendix H, "Assumptions for Evaluating the Radiological Consequences of a Pressurized-Water Reactor Control Rod Ejection Accident," considers two potential release paths to the environment. One pathway is via the secondary side through the steam generator power operated relief valves (PORVs). This pathway assumes that all of the released activity remains in the RCS and that due to primary-to-secondary leakage releases occur via the secondary system. For this case, no releases from the containment are assumed.

The second pathway considers a release into containment and a subsequent release to the environment via containment leakage. This pathway postulates that the rod ejection creates a breach in the RCS. For this pathway, 100% of the activity released from the fuel is assumed to be released into the containment. Depending on the size of the breach, this could result in the pressurization of the containment above normal operating conditions due to what would essentially be a small to medium size LOCA. Therefore, Appendix H of RG 1.183 specifies that releases due to containment leakage are also to be modeled for an REA. These two pathways are to be analyzed separately, and there is no summation of the dose results.

Consistent with UFSAR Section 15.4.3.1, the licensee concluded that analysis of the containment leak path for the SRW accident was unnecessary because primary coolant boundary is expected to remain intact and the only leakage from the RCS to containment would be minor and would not result in a pressurization of the containment. Therefore, any release from containment would be insignificant. Consequently, the licensee assumed that the only pathway necessary for consideration for release to the environment would be the primary-to-secondary leak path and the resultant release via steaming through the steam generator PORVs. The licensee made no changes to the UFSAR Section 15.4.3.1 assumptions concerning the steam releases for the SRW accident.

3.2.4.3 Control Room Habitability for the Single Rod Withdrawal Accident

The licensee changed the UFSAR Section 15.4.3.1 assumptions by choosing not to credit the CR emergency filtration system for the analysis of the CR dose resulting from a postulated SRW accident. The licensee maintained the assumptions for the normal outside air makeup flow rate of 400 cfm and an additional 230 cfm of unfiltered CR inleakage. Therefore, the licensee maintained a total of 630 cfm of unfiltered airflow into the CR for the duration of the analysis period. Not crediting the CR emergency filtration system for the analysis of the CR dose resulting from a postulated SRW accident is conservative and, therefore, acceptable to the NRC staff.

3.2.4.4 Single Rod Withdrawal Accident Conclusion

The licensee evaluated the radiological consequences resulting from the postulated SRW accident and concluded that the radiological consequences at the EAB, LPZ, and CR are within the dose acceptance criteria specified in 10 CFR 50.67 and the accident dose acceptance

criteria specified in SRP Section 15.0.1. The NRC staff found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE. The assumptions the NRC staff found acceptable are in Table 8 of the Attachment to this SE and the licensee's calculated dose results are in Table 1 of the Attachment to this SE. The NRC staff performed independent sensitivity dose evaluations as necessary to ensure a thorough understanding of the licensee's methods. The NRC staff finds that the EAB, LPZ, and CR doses estimated by the licensee for the LRA meet the applicable accident dose acceptance criteria and are, therefore, acceptable.

3.2.5 Main Steam Line Break Accident

The postulated MSLB accident assumes a break of one main steam line outside the primary containment. This leads to an uncontrolled release of steam from the steam system. The resultant depressurization of the steam system causes the main steam isolation valves to close and causes the reactor to trip. For the MSLB DBA radiological consequence analysis, a LOOP is assumed to occur shortly after the trip signal. Following a reactor trip and turbine trip, the radioactivity is released to the environment through the steam generator PORVs. Because the LOOP renders the main condenser unavailable, the plant is cooled down by releasing steam to the environment.

The radiological consequences of a main steam line break outside containment will bound the consequences of a break inside containment. Therefore, only the MSLB outside of containment is considered with regard to the radiological consequences. The affected steam generator, hereafter referred to as the faulted steam generator, rapidly depressurizes and releases the initial contents of the steam generator to the environment. The MSLB accident is described in UFSAR Section 15.1.5, "Main Steamline Break Event." RG 1.183, Appendix E, Assumptions for Evaluating the Radiological Consequences of a PWR Main Steam Line Break Accident," identifies acceptable radiological analysis assumptions for a PWR MSLB accident.

The steam release from a rupture of a main steam line would result in an initial increase in steam flow, which decreases during the accident as the steam pressure decreases. The increased energy removal from the RCS causes a reduction of coolant temperature and pressure. In the presence of a negative moderator temperature coefficient, the cooldown results in an insertion of positive reactivity. If the most reactive control rod assembly is assumed stuck in its fully withdrawn position after the reactor trip, there is an increased possibility that the core will become critical and return to power. A return to power following a steam line rupture could be a potential problem because of the high hot channel factors that may exist when the most reactive control rod is assumed stuck in its fully withdrawn position. The core power transient is stabilized by Doppler feedback and by the decrease in moderator density in the core. With time, the increasing boron concentration in the reactor causes the power to continuously reduce. When the operator secures auxiliary feedwater, the temperature of the RCS increases and the event is terminated.

3.2.5.1 Source Term

As documented in the SE for HBRSEP Amendment No. 201 issued September 24, 2004 (Reference 8), in UFSAR Section 15.1.5 the licensee evaluated the consequences of an MSLB accident for three cases. The first case assumed the MSLB occurred following an iodine spike, referred to as the pre-existing spike case. The second case assumed the MSLB initiated an iodine spike, referred to as the accident-initiated spike. The third case assumed that the MSLB induced fuel failures resulting in the release of the gap activity from two breached fuel

assemblies. For this LAR the licensee only included an evaluation of the fuel breach case since the source term for this evaluation changed as a result of the 24 month fuel cycle. The cases based on RCS activity are controlled by TSs on RCS activity and are not impacted by the extended fuel cycle. In addition the fuel breach case is the most limiting.

RG 1.183, Appendix E, Regulatory Position 1, states, "Assumptions acceptable to the NRC staff regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this regulatory guide. The release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached. The fuel damage estimate should assume that the highest worth control rod is stuck at its fully withdrawn position." For the fuel breach case, the licensee assumed that two fuel assemblies were breached releasing the activity in the fuel gap and that these assemblies had a maximum radial peaking factor of 1.8.

For the MSLB accident, the licensee evaluated the radiological dose contribution from the release of secondary side activity using the TS 3.7.15, "Secondary Specific Activity," limiting condition for operation (LCO) of 0.10 microcurie/gram (μCi/gm) dose equivalent iodine 131 (DEI).

The NRC staff finds that the licensee incorporated the following regulatory positions in the evaluation of the MSLB accident:

> RG 1.183, Appendix E, Regulatory Position 4, which states that, "The chemical form of radioiodine released from the fuel should be assumed to be 95% cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. Iodine releases from the steam generators to the environment should be assumed to be 97% elemental and 3% organic. These fractions apply to iodine released as a result of fuel damage and to iodine released during normal operations, including iodine spiking."
>
> RG 1.183, Appendix E, Regulatory Position 5.5.4, which states that, "The radioactivity in the bulk water is assumed to become vapor at a rate that is the function of the steaming rate and the partition coefficient. A partition coefficient for iodine of 100 may be assumed. The retention of particulate radionuclides in the steam generators is limited by the moisture carryover from the steam generators."

3.2.5.2 Transport

UFSAR Section 15.1.5.4, "Radiological Consequences," describes the assumptions regarding primary-to-secondary leak rates in the steam generators as follows:

> The primary-to-secondary leak rate in the steam generators is based on the leak-rate-limiting condition for operation specified in the Technical Specifications (150 gpd, which is 0.104 gpm [gallon per minute]). The leakage is apportioned between the steam generators in such a manner that the calculated dose is maximized. The operational primary-to-secondary leakage is conservatively assumed to be 0.11 gpm through any one SG [steam generator] and 0.3 gpm total to all three SGs. Since the tube leak into the faulted SG and subsequently to the environment continues until the RCS temperature drops below 212 °F at 98.8 hours, it is conservative to assign the maximum allowed 0.11 gpm to the faulted SG with the remainder of 0.19 gpm assigned to the unaffected SGs.

For the MSLB accident analysis in its LAR, the licensee clarified the total primary-to-secondary leakage rate from 0.3 gpm (432 gpd) to 0.3125 gpm (450 gpd). The proposed value of 450 gpd clearly reflects the total accident induced leakage conditions relative to the TS 3.4.13, "RCS Operational Leakage," LCO, which states that RCS operational primary-to-secondary leakage shall be limited to 75 gpd through any one steam generator. In addition, for this LAR the licensee assigned leakage to the faulted steam generator of 0.115 gpm (165.6 gpd) and 0.1975 gpm (284.4 gpd) to the two unaffected steam generators. Therefore, the licensee proposes to modify the above description in UFSAR Section 15.1.5.4 as follows:

> The primary-to-secondary leak rate in the steam generators is based on the maximum accident induced steam generator tube leakage criterion in the Technical Specifications of 150 gpd (0.104 gpm) per steam generator for a total of 450 gpd (0.3125 gpm). The analysis conservatively assigns leakage to the faulted steam generator of 0.115 gpm and 0.1975 gpm to the unaffected generators.

RG 1.183, Appendix E, Regulatory Position 5.3, states that, "The primary-to-secondary leakage should be assumed to continue until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 100°C (212°F). The release of radioactivity from unaffected steam generators should be assumed to continue until shutdown cooling is in operation and releases from the steam generators have been terminated." Consistent with UFSAR Section 15.1.5, the licensee assumed that primary-to-secondary leakage continues until the RCS temperature drops below 212 °F at 98.8 hours. The licensee made no changes to the UFSAR Section 15.1.5 assumptions concerning the steam releases for the MSLB accident evaluation.

In accordance with RG 1.183 and UFSAR Section 15.1.5, the licensee assumed that all noble gas radionuclides released from the primary system are released to the environment without reduction or mitigation. Following the guidance from RG 1.183, Appendix E, Regulatory Positions 5.5.1, 5.5.2 and 5.5.3 and UFSAR Section 15.1.5, the licensee assumed that all of the primary-to-secondary leakage into the faulted steam generator will flash to vapor, and be released to the environment with no mitigation. For the unaffected steam generators that are used for plant cooldown, the licensee assumed that the primary-to-secondary leakage mixes with the secondary water without flashing.

RG 1.183, Appendix E, Regulatory Position 5.5.4, states that, "The radioactivity in the bulk water is assumed to become vapor at a rate that is the function of the steaming rate and the partition coefficient. A partition coefficient for iodine of 100 may be assumed. The retention of particulate radionuclides in the steam generators is limited by the moisture carryover from the steam generators."

Consistent with RG 1.183 and UFSAR Section 15.1.5, the licensee assumed that the radioactivity in the bulk water of the unaffected steam generators becomes vapor at a rate that is a function of the steaming rate and the partition coefficient. The licensee used a partition coefficient of 100 for elemental iodine and other particulate radionuclides released from the intact steam generators.

3.2.5.3 Control Room Ventilation Assumptions for the Main Steam Line Break

The NRC staff notes that the licensee did not change any of the UFSAR Section 15.1.5 CR ventilation assumptions regarding the MSLB accident dose consequence analysis for this LAR. Consistent with UFSAR Section 15.1.5, the licensee assumed that the normal ventilation system

would be operating for 50 seconds post-MSLB prior to the automatic initiation of the CR's emergency ventilation system. During this period of normal operation, in addition to the 400 cfm of normal unfiltered intake flow, the licensee assumed an unfiltered inleakage rate into the CR of 300 cfm. After the safety injection signal initiates the CR emergency ventilation system the intake flow of 400 cfm is directed to the CR emergency filtration system that is comprised of both particulate filters and a charcoal adsorber. In addition, the CR emergency filtration system recirculates CR air at a rate of 2,600 cfm.

In accordance with RG 1.183, the licensee analyzed the dose consequences of the MSLB accident assuming a LOOP. With offsite power available, the Hagan Room, which is adjacent to the CR, remains at a negative pressure relative to the CR due to the operation of the auxiliary building exhaust fan HVE-7. The UFSAR Section 15.1.5 CR dose consequence analysis assumes that, as a result of the LOOP and the loss of HVE-7, for the first hour the Hagan Room would be at a higher pressure than the CR resulting in an increased CR unfiltered inleakage of 300 cfm. The Hagan Room is assumed to remain in this condition until operators can take manual actions that will result in the reduction of the pressure in the Hagan Room to below that of the CR. The licensee's analysis assumed that it would take approximately 1 hour to implement the manual actions to reduce the pressure in the Hagan Room. Therefore, the UFSAR Section 15.1.5 MSLB accident CR dose consequence analysis assumes that for the first hour of the accident the CR unfiltered inleakage rate is 300 cfm. After 1 hour, the unfiltered inleakage is assumed to be reduced by 70 cfm to 230 cfm when the inleakage from the Hagan Room is reduced.

The NRC staff notes that the letter dated April 29, 2014 (Reference 10), documents, as discussed above, that based on CR inleakage testing in 2003 and 2012, the licensee determined that the actual inleakage to the CR if no manual action is taken (and the associated 1-hour delay) is less than the assumed inleakage to CR after 1 hour. Thus, using a higher inleakage for the first hour of a CR dose analysis is conservative relative to measured values. Therefore maintaining the UFSAR Section 15.1.5 assumption of a 1-hour delay before the reduction in CR unfiltered inleakage from 300 cfm to 230 cfm is conservative for the analysis of CR habitability regarding the MSLB accident.

The NRC staff further notes that CR unfiltered inleakage testing supports the lower CR infiltration assumptions (170 and 100 cfm) used in the LOCA dose consequence analysis. Therefore, the use of higher values of CR unfiltered inleakage for other DBA CR dose analyses (300 and 230 cfm) is an additional conservatism for these analyses.

3.2.5.4 Main Steam Line Break Accident Conclusions

The licensee evaluated the radiological consequences resulting from the postulated MSLB accident and concluded that the radiological consequences at the EAB, LPZ, and CR are within the dose acceptance criteria specified in 10 CFR 50.67 and the accident specific dose criteria specified in SRP Section 15.0.1. The NRC staff finds that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE. The assumptions the NRC staff finds acceptable are in Table 9 of the Attachment to this SE and the licensee's calculated dose results are in Table 1 of the Attachment to this SE. The NRC staff performed independent sensitivity dose evaluations as necessary to ensure a thorough understanding of the licensee's methods. The staff finds that the EAB, LPZ, and CR doses estimated by the licensee for the MSLB accident meet the applicable accident dose acceptance criteria and are therefore acceptable.

3.2.6 Rod Ejection Accident

UFSAR Section 15.4.8, "Spectrum of Rod Cluster Control Assembly (RCCA) Ejection Accidents," describes the REA accident as the mechanical failure of a control rod mechanical pressure housing such that the coolant system pressure ejects an RCCA and drive shaft to a fully withdrawn position. The consequences of this mechanical failure are a rapid reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage. Following the guidance in RG 1.183, Appendix H, the licensee evaluated two separate release scenarios for the REA. In the first case, the REA is assumed to induce a LOCA resulting in a release of fission products into the containment atmosphere and a subsequent release to the environment from the containment leakage pathway.

For the second case, the radiological consequences from a REA are evaluated assuming that the RCS boundary remains intact and that fission products are released to the environment from the secondary system. In this case, fission products from the damaged fuel are assumed to be released to the primary coolant and transported to the secondary system through primary-to-secondary leakage in the steam generators. The REA accident is analyzed with the assumption of a concurrent LOOP that causes steam releases from the secondary system to occur through the steam generator PORVs and safety valves to the environment.

The licensee determined that the containment release scenario is the limiting case for an REA at HBRSEP, therefore the REA doses shown in Table 1 of the Attachment to this SE are for the containment release scenario.

3.2.6.1 Source Term

The source term for the REA is based on the assumption that 10% of the rods in the core experience DNB resulting in damage to the fuel cladding but that no fuel melt occurs as a result of the REA. Consistent with the guidance provided in RG 1.183, Appendix H, the licensee assumed that 10% of the core inventory of noble gases and iodine reside in the fuel gap. In addition, the licensee assumed that 10% of core inventory of bromine and 12% of the core inventory of alkali metals reside in the fuel gap and are included in the source term. The licensee assumed that for the 10% of the fuel that experiences DNB, all of the gap activity contained in the affected fuel will be available for release in both the REA induced LOCA scenario and the secondary side release scenario. The licensee applied a radial peaking factor of 1.8 to the REA source term.

In addition to the source term resulting from the 10% of rods experiencing DNB, the licensee included the contribution from the initial RCS specific activity concentration at 0.5 μCi/gm DEI as well as the contribution from the release of secondary coolant at a concentration of 0.1 μCi/gm DEI.

3.2.6.2 Transport from Containment

In accordance with RG 1.183, Appendix H, Regulatory Position 3, the licensee assumed that 100% of the released activity is released instantaneously and mixed homogeneously throughout the containment atmosphere for the REA induced LOCA scenario. The licensee assumed that the activity released to the containment through the rupture in the reactor vessel head mixes instantaneously throughout the containment. The licensee did not credit the removal of iodine in the containment due to the operation of containment sprays. The licensee applied a natural aerosol deposition coefficient of 0.1 per hour to credit the natural deposition of particulate iodine

in the containment. The licensee assumed a containment leakage rate of 0.1 weight percent per day for the first 24 hours and 0.05 weight percent per day thereafter.

3.2.6.3 Transport from Secondary System

In accordance with RG 1.183, Appendix H, Regulatory Position 3, the licensee assumed that 100% of the released activity is released instantaneously and completely dissolved in the primary coolant in the REA secondary side release scenario. In accordance with RG 1.183, Appendix H, Regulatory Position 7, the licensee evaluated the transport of activity from the RCS to the steam generators secondary side assuming a total primary-to-secondary leak rate equal to 450 gpd.

In accordance with RG 1.183 the licensee included a partition coefficient of 100 for halogens and alkali metal releases from the steam generators. Because of their volatility, 100% of the noble gases are assumed to be released without holdup, reduction or mitigation. The licensee assumed that the steaming release from either the steam generator atmospheric relief valves or safety valves ends after 53.2 hours, which is the time required for one train of the RHR System to establish adequate shutdown cooling to terminate releases from the steam generators.

3.2.6.4 Control Room Ventilation Assumptions for the Rod Ejection Accident

For the coolant release scenario, the licensee assumed that the CR emergency filtration system will become operational at 1 hour following the onset of the accident. Prior to the initiation of the CR emergency ventilation system 400 cfm of unfiltered intake air enters the CR through the normal ventilation system unfiltered. In addition, prior to the initiation of the CR emergency ventilation, the licensee assumed an unfiltered inleakage of 300 cfm. After the initiation of the CR emergency ventilation system the licensee modeled the CR ventilation assuming 400 cfm of filtered pressurization intake as well as 2600 cfm of filtered recirculation. During the operation of the CR emergency ventilation system, unfiltered inleakage into the CR was assumed to be 230 cfm.

For the containment release scenario, the licensee credited the CR emergency filtration system initiation at 335 seconds after the accident which includes a 5-minute delay added to the 35 seconds for the safety injection signal response time. Prior to the initiation of the CR emergency ventilation system 400 cfm of unfiltered intake air enters the CR through the normal ventilation system unfiltered. In addition, prior to the initiation of the CR emergency ventilation, the licensee assumed an unfiltered inleakage of 300 cfm. After the initiation of the CR emergency ventilation system, the licensee modeled the CR ventilation assuming 400 cfm of filtered pressurization intake as well as 2600 cfm of filtered recirculation. During the operation of the CR emergency ventilation system, unfiltered inleakage into the CR was assumed to be 230 cfm.

The NRC staff notes that the letter dated April 29, 2014 (Reference 10), documents that based on CR inleakage testing in 2003 and 2012, the licensee determined that the actual inleakage to the CR if no manual action is taken (and the associated 1-hour delay) is less than the assumed inleakage to CR after 1 hour. Thus, using a higher inleakage for the first hour of a CR dose analysis is conservative relative to measured values. Therefore for the REA, which had no UFSAR Section 15.4.8 dose consequence analysis, the licensee did not incorporate the 1-hour delay and reduced the initial assumed unfiltered inleakage of 300 cfm would be reduced to an assumed value of 230 cfm in conjunction with the initiation of the CR emergency filtration system.

The NRC staff further notes that CR unfiltered inleakage testing supports the lower CR infiltration assumptions (170 and 100 cfm) used in the LOCA dose consequence analysis. Therefore, the use of higher values of CR unfiltered inleakage (300 and 230 cfm) for other DBA CR dose analyses such as the REA is an additional conservatism.

3.2.6.5 Rod Ejection Accident Conclusions

The licensee evaluated the radiological consequences resulting from the postulated REA and concluded that the radiological consequences at the EAB, LPZ, and CR are within the dose acceptance criteria specified in 10 CFR 50.67 and the accident specific dose criteria specified in SRP Section 15.0.1. The NRC staff concluded that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE. The assumptions NRC staff found acceptable are in Table 10 of the Attachment to this SE and the licensee's calculated dose results for the limiting containment release scenario are in Table 1 of the Attachment to this SE. The NRC staff performed independent sensitivity dose evaluations as necessary to ensure a thorough understanding of the licensee's methods. The EAB, LPZ, and CR doses estimated by the licensee for the REA meet the applicable accident dose acceptance criteria and are, therefore, acceptable.

3.3 Atmospheric Dispersion Estimates

3.3.1 Meteorological Data

The UFSAR Chapter 15.0, "Accident Analysis," atmospheric dispersion factor (χ/Q) values for the DBA dose assessments described above were determined using onsite meteorological data collected during calendar years 1988 through 1996. These data were previously approved by the NRC staff by HBRSEP Amendment No. 195, issued October 4, 2002 (Reference 7).

3.3.2 Control Room Atmospheric Dispersion Factors

As described in the SEs associated with HBRSEP Amendments Nos. 195 and 201, issued October 4, 2002 (Reference 7) and September 24, 2004 (Reference 8), respectively, the licensee used the ARCON96 methodology, as described in NUREG/CR-6331, Revision 1, "Atmospheric Relative Concentrations in Building Wake" (Reference 22), for calculation of CR χ/Q values with a modification to the surface roughness length and averaging sector width constant. These two modifications are acceptable to the NRC staff. Calculations were made for postulated DBA releases to the CR from the plant stack, closest main steam safety valve/relief valve, closest main steam line, nearest point of the containment building, and RHR heat exchanger room, and to the technical support center and emergency operations facility from the nearest point of the containment building and RHR heat exchanger room. All releases were assumed to be ground-level point releases. The licensee made no changes to the previously approved UFSAR Chapter 15.0 CR χ/Q values for this LAR.

3.3.3 Offsite Atmospheric Dispersion Factors

As described in the SE issued with the Amendment No. 195, issued October 4, 2002 (Reference 7), the licensee calculated χ/Q values for the EAB and LPZ using site-specific inputs and the PAVAN computer code. The PAVAN code, documented in NUREG/CR-2858, "PAVAN: An Atmospheric Dispersion Program for Evaluating Design Basis Accidental Releases of Radioactive Materials from Nuclear Power Plants" (Reference 23) uses the methodology

described in RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants" (Reference 24). The licensee made calculations for an EAB distance of 425 meters and LPZ distance of 7242 meters. Releases were assumed to be ground level. The licensee made no changes to the previously approved UFSAR Chapter 15.0 offsite atmospheric dispersion factors (χ/Qs) for this LAR.

3.4 Licensing Bases Changes

The licensee stated that this license amendment will require changes to the TS Bases, which are controlled by TS 5.5.14, "Technical Specifications (TS) Bases Control Program," the UFSAR, and the TRM. The changes to the TS Bases and TRM were provided for information purposes and were not reviewed by the NRC staff. By letter dated April 3, 2017, the licensee separately submitted a LAR requesting TS changes for the extended fuel cycle (Reference 3).

3.5 Commitments

This LAR to modify the licensing basis AST contains no new regulatory commitments.

3.6 Staff Evaluation Conclusion

As described above, the NRC staff reviewed the assumptions, inputs, and methods used by the licensee to assess the radiological consequences of DBAs. The NRC staff finds that the licensee used analysis methods and assumptions consistent with the regulatory requirements and guidance identified in Section 2.0 above. The NRC staff compared the doses estimated by the licensee to the applicable acceptance criteria identified in Section 2.0. The NRC staff finds there is reasonable assurance that the licensee's estimates of the EAB, LPZ, and CR doses will comply with these criteria, and that HBRSEP will continue to provide sufficient safety margins with adequate defense-in-depth to address unanticipated events and to compensate for uncertainties in accident progression and analysis assumptions and parameters. Therefore, the NRC staff concludes that the proposed modifications to AST implementation in UFSAR Chapter 15.0 are acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the State of South Carolina official was notified of the proposed issuance of the amendment on August 4, 2017. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding published in the *Federal Register* on November 22, 2016 (81 FR 83875). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9).

Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that 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.

7.0 REFERENCES

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2. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000 (ADAMS Accession No. ML003716792).

3. Kapopoulous, E. J. Jr., Duke Energy Progress, Inc., letter to U.S. Nuclear Regulatory Commission, "Resubmittal of Request for Technical Specification Change to Change Technical Specification Surveillance Requirement Frequencies to Support 24-Month Fuel Cycles," April 3, 2017 (ADAMS Accession No. ML17093A787)

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10. Peavyhouse, S. W., Duke Energy Progress, Inc., letter to U.S. Nuclear Regulatory Commission, "Report of Changes Pursuant to 10 CFR 50.59," April 29, 2014 (ADAMS Accession No. ML14133A007).

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12. U.S. Environmental Protection Agency, EPA-520-1-88-020, "Federal Guidance Report No. 11 - Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," September 1988 (ADAMS Accession No. ML101590171).

13. U.S. Environmental Protection Agency, EPA-402-R-93-081, "Federal Guidance Report No. 12 - External Exposure to Radionuclides in Air, Water, and Soil," September 1993 (ADAMS Accession No. ML101590169).

14. Repko, R. T., Duke Energy Carolinas, LLC, letter to U.S. Nuclear Regulatory Commission, "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 (ADAMS Accession No. ML15196A093).

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Attachment: Accident Consequences Data
and Assumptions, Tables 1 - 10

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Date: September 29, 2017

Table 1
Radiological Consequences Expressed as TEDE [1] (rem)

Design Basis Accidents	EAB [2]	LPZ [3]	CR
Loss of Coolant Accident	22.5	1.53	4.40
Main Steamline Break Accident [4]	1.72	0.38	1.53
Dose acceptance criteria	25	25	5
Rod Ejection Accident	4.07	0.69	4.83
Fuel Handling Accident Inside Containment	4.34	0.22	3.56
Fuel Handling Accident Inside FHB	1.46	0.07	0.33
Dose acceptance criteria	6.3	6.3	5
Single Rod Withdrawal	0.69	0.05	2.57
Locked Rotor Accident	2.09	0.16	0.68
Dose acceptance criteria	2.5	2.5	5

[1] Total effective dose equivalent
[2] Exclusion area boundary
[3] Low population zone
[4] MSLB assumes fuel failure

Table 2
Control Room Atmospheric Dispersion Factors

Source Location / Duration		χ/Q (sec/m^3) [1]
For evaluating releases from containment leakage		
	0 - 2 hours	4.15×10^{-3}
	2 - 8 hours	2.74×10^{-3}
	8 - 24 hours	1.17×10^{-3}
	24 - 96 hours	8.18×10^{-4}
	96 - 720 hours	6.74×10^{-4}
For evaluating releases from the RHR Heat Exchanger Room		
	0 - 2 hours	7.13×10^{-3}
	2 - 8 hours	5.49×10^{-3}
	8 - 24 hours	2.29×10^{-3}
	24 - 96 hours	1.71×10^{-3}
	96 - 720 hours	1.37×10^{-3}
For evaluating releases from inside the FHB		
	0 - 2 hours	1.24×10^{-3}
	2 - 8 hours	8.97×10^{-4}
	8 - 24 hours	3.62×10^{-4}
	24 - 96 hours	2.58×10^{-4}
	96 - 720 hours	2.14×10^{-4}
For evaluating releases from MSSV[2] /PORVs		
	0 - 2 hours	2.60×10^{-3}
	2 - 8 hours	1.65×10^{-3}
	8 - 24 hours	7.22×10^{-4}
	24 - 96 hours	4.97×10^{-4}
	96 - 720 hours	4.01×10^{-4}
For evaluating releases from MSLs[3]		
	0 - 2 hours	2.48×10^{-3}
	2 - 8 hours	1.57×10^{-3}
	8 - 24 hours	7.05×10^{-4}
	24 - 96 hours	4.74×10^{-4}
	96 - 720 hours	3.93×10^{-4}

[1] seconds per cubic meter
[2] main steam safety valve
[3] main steam lines

Table 3
Offsite Atmospheric Dispersion Factors (sec/m^3)

Receptor / Duration		χ/Q (sec/m^3)
EAB		
	0 - 2 hours	1.77×10^{-3}
	2 - 8 hours	1.77×10^{-3}
	8 - 24 hours	1.77×10^{-3}
	24 - 96 hours	1.77×10^{-3}
	96 - 720 hours	1.77×10^{-3}
LPZ		
	0 - 2 hours	8.92×10^{-5}
	2 - 8 hours	3.50×10^{-5}
	8 - 24 hours	2.19×10^{-5}
	24 - 96 hours	7.95×10^{-6}
	96 - 720 hours	1.85×10^{-6}

Table 4
Control Room Data and Assumptions

Control room (CR) volume	20,124 ft^3
CR outside air make up	400 cfm [1]
CR normal mode intake filter efficiencies	0% (all species)
CR ventilation emergency mode start times[2]	
LOCA on SI Signal	35 seconds
MSLB on SI Signal	50 seconds
REA Containment release scenario	5 minutes 35 seconds
FHA, LRA, & REA secondary release scenario	1 hour
CR ventilation emergency mode recirculation flow rate	2600 cfm
CR emergency mode credited intake filter efficiency[3]	
Elemental iodine	95%
Organic iodide	95%
Particulates	99%
CR assumptions for Normal Mode unfiltered inleakage	
LOCA	170 cfm
FHA, LRA, SRW, MSLB, REA	300 cfm
CR assumptions for Emergency Mode unfiltered inleakage	
LOCA after 1 hour	100 cfm
MSLB from 50 seconds to 1 hour	300 cfm
FHA, LRA, MSLB after 1 hour	230 cfm
REA after 335 seconds containment release scenario	230 cfm
REA after 1 hour secondary release scenario	230 cfm
CR operator breathing rate (0 – 720 hours)	3.5×10^{-4} m^3/sec
CR occupancy factors	
0 - 24 hours	1.0
24 - 96 hours	0.6
96 - 720 hours	0.4

[1] cubic feet per minute

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[2] The licensee did not credit the CR emergency filtration system for the SRW dose consequence analysis
[3] In LAR, Enclosure II, Table 6, for the REA, the licensee inadvertently mislabeled the CR ventilation emergency mode Iodine intake filter efficiencies as CR ventilation normal mode Iodine intake filter efficiencies.

Table 5
Data and Assumptions for the LOCA

Reactor power with uncertainty	2346 MWth [1]	
Containment volume	1,958,526 ft^3	
Containment leak rate		
0 - 24 hours	0.1 percent weight/day	
24 - 720 hours	0.05 percent weight/day	
Containment volume sprayed region (B train)	1,596,198.7 ft^3	
Containment volume unsprayed region	362,327.3 ft^3	
Containment cooling fans combined start time	76 seconds	
Containment cooling fans combined flow rate	65,000 cfm	
Elemental iodine removal coefficients (spray train B)	Sprayed region	Unsprayed
0 – 3 min (spray auto initiation @ 3 min)	0.0 hr^{-1}	0.0 hr^{-1}
3 – 77 min (spray stopped for switchover @ 77 min)	20.0 hr^{-1}	0.0 hr^{-1}
77 – 87 minutes (10 min for switch to recirculation)	0.0 hr^{-1}	0.0 hr^{-1}
87 min – 2.04 hours (DF of 200 @ 2.04 hours)	20.0 hr^{-1}	0.0 hr^{-1}
2.04 - 720 hours	0.0 hr^{-1}	0.0 hr^{-1}
Aerosol removal coefficients (spray train B)	Sprayed region	Unsprayed
0 – 3 min (spray auto initiation @ 3 min)	0.1 hr^{-1}	0.1 hr^{-1}
3 – 77 min (spray stopped for switchover @ 77 min)	3.427 hr^{-1}	0.1 hr^{-1}
77 – 87 minutes (10 min for switch to recirculation)	0.1 hr^{-1}	0.1 hr^{-1}
1.45 – 2.76 hours (@ 2.76 hr DF = 50)	3.427 hr^{-1}	0.1 hr^{-1}
2.76 – 2.78 hours (spray terminated @ 2.78 hours)	0.3427 hr^{-1}	0.1 hr^{-1}
2.78 – 720 hours	0.1 hr^{-1}	0.1 hr^{-1}
Assumptions for the dose consequence from ESF leakage[4]		
Assumed ESF leakage (two times TRM limit)	2 gph[2]	
Assumed ECCS leakage start time	21 minutes	
ECCS Flashing fraction	10%	
Containment sump pH	>7	
Volume of water in containment sump		
From 0 to 21 minutes	N/A (recirc. starts at 21 minutes)	
From 21 minutes to 40 minutes	35,850 ft^3	
From 40 minutes to 51.5 minutes	40,889 ft^3	
From 51.5 minutes to 30 days	43,939 ft^3	

[1] Megawatt thermal
[2] gallon per hour

[4] Due to the location of the RWST with respect to the CR, for conservative reasons, the licensee maintained the CLB assumption that ESF system leakage is from the RHR heat exchanger room (located in the Auxiliary Building), which has a greater atmospheric dispersion factor.

Table 6
Data and Assumptions for the FHA

Reactor power with uncertainty	2346 MWth	
Minimum post shutdown fuel handling time (decay time)	116 hours	
Radial peaking factor	1.8	
Number of assemblies damaged	1	
Number of fuel assemblies in the core	157	
Total number of fuel rods per assembly	204	
Number of rods that can exceed 6.3 kW/ft	35	
Fuel clad damage gap release fractions	≤6.3 kW/ft	>6.3 kW/ft
^{131}I	0.08	0.08
^{85}Kr	0.1	0.30
Remainder of halogens and noble gases	0.05	0.05
^{134}Cs and ^{137}Cs (Alkali metals)	0.12	0.36
Other Alkali metals	0.12	0.12
Minimum fuel pool water depth	21 feet	
Fuel pool DF for FHA in fuel handling building		
Noble gases and organic iodine	1	
Aerosols	Infinite	
Elemental iodine (21 ft of water cover)	174	
Overall iodine (21 ft of water cover)	138 (effective DF)	
Minimum reactor cavity water depth	23 feet	
Reactor cavity DF for FHA in containment		
Noble gases and organic iodine	1	
Aerosols	Infinite	
Elemental iodine (23 ft of water cover)	285	
Overall iodine (23 ft of water cover)	200 (effective DF)	
Duration of release to the environment	2 hours	
Release location for FHA in fuel handling building	Plant vent	
Release location for FHA in containment	Closest from containment to CR	
Filtration efficiencies	Fuel Building	CR
Elemental	90%	95%
Particulate	99%	99%
Organic	70%	95%

Table 7
Data and Assumptions for the Locked Rotor Accident

Reactor power with uncertainty	2346 MWt
Number of assemblies experiencing DNB	12
Radial peaking factor	1.8
Number of rods that can exceed 6.3 kW/ft and experience DNB	0
Time to establish shutdown cooling and terminate release	53.2 hours
RCS specific activity	0.5 μCi/gm DEI
Secondary side specific activity	0.1 μCi/gm DEI
RCS volume	8,254 ft^3
Secondary side volume	4,262 ft^3
Primary to secondary leak rate	450 gpd
Steam release from steam generators	
0 – 2 hours	301,967.3 lbm[1]
2 – 8 hours	566,768.3 lbm
8 – 24 hours	1,124,995.8 lbm
24 – 53.2 hours	1,637,910.1 lbm

[1] pound-mass

Table 8
Data and Assumptions for the SRW Accident

Reactor power with uncertainty	2346 MWt
Number of assemblies experiencing DNB	4
Number of assemblies experiencing fuel melt	0
Radial peaking factor	1.8
Number of rods that can exceed 6.3 kW/ft and experience DNB	0
Time to establish shutdown cooling and terminate release	53.2 hours
RCS specific activity	0.5 µCi/gm DEI
Secondary side specific activity	0.1 µCi/gm DEI
RCS volume	8,254 ft^3
Secondary side volume	4,262 ft^3
Primary to secondary leak rate	450 gpd
Steam release from steam generators	
0 – 2 hours	301,967.3 lbm
2 – 8 hours	566,768.3 lbm
8 – 24 hours	1,124,995.8 lbm
24 – 53.2 hours	1,637,910.1 lbm

Table 9
Data and Assumptions for the MSLB Accident

Reactor power with uncertainty	2346 MWt
Number of damaged assemblies	2
Radial peaking factor	1.8
Number of rods that can exceed 6.3 kW/ft and experience DNB	0
RCS volume	8,254 ft^3
RCS specific activity	0.5 µCi/gm DEI
Secondary side volume	4,262 ft^3
Secondary side specific activity	0.1 µCi/gm DEI
Primary to secondary leak rate	450 gpd
Duration of steaming releases from unaffected steam generators	53.2 hours
Duration of releases from faulted steam generator (Time when RCS Temp.<212°F)	98.8 hours

Steam release from Faulted Steam Generator	Interval (lbm)	Integrated (lbm)
At time of break	161,194	161,194
0 - 2 hours	110.2	161,304.2
2 - 8 hours	330.5	161,634.7
8 - 24 hours	881.3	162,516.0
24 - 53.2 hours	1,608.3	164,124.3
53.2 - 98.8 hours	2,511.8	166,636.1
Steam release from unaffected Steam Generator	**Interval (lbm)**	**Integrated (lbm)**
At time of break	0	0
0 - 2 hours	300,116.1	300,116.1
2 - 8 hours	561,234.8	861,350.9
8 - 24 hours	1,110,326.4	1,971,677.3
24 - 53.2 hours	1,611,091.5	3,582,768.8
53.2 - 98.8 hours	0	N/A

Table 10
Data and Assumptions for the REA

Reactor power with uncertainty	2346 MWt
Percentage of rods experiencing DNB	10%
Radial peaking factor	1.8
Number of rods that can exceed 6.3 kW/ft and experience DNB	0
Time to establish shutdown cooling and terminate release	53.2 hours
Initial RCS specific activity	0.5 µCi/gm DEI
Initial secondary side activity concentration	0.1 µCi/gm DEI
Containment volume	1,958,526 ft^3
Containment cooling fans start time	76 seconds
Containment cooling fans combined flow rate	65,000 cfm
Natural deposition removal coefficient	0.1 hr^{-1}
Containment leak rate	
First 24 hours	0.1 weight percent per day
Remainder of analysis period	0.05 weight percent per day
Minimum RCS volume (574.9 °F and 2235 psig)	8,254 ft^3
Secondary side volume	4,262 ft^3
Total primary-to-secondary leak rate	450 gpd
Steam release from steam generators	
0 – 2 hours	301,967.3 lbm
2 – 8 hours	566,768.3 lbm
8 – 24 hours	1,124,995.8 lbm
24 – 53.2 hours	1,637,910.1 lbm
Steam Generator partition coefficient halogens and alkali metals	100

SUBJECT: H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2 – ISSUANCE OF AMENDMENT REGARDING REQUEST TO MODIFY THE LICENSING BASIS ALTERNATE SOURCE TERM (CAC NO. MF8378) DATED SEPTEMBER 29, 2017

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