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

March 30, 2015

Mr. Ernest J. Harkness
Site Vice President
FirstEnergy Nuclear Operating Company
Mail Stop A-PY-A290
P.O. Box 97, 10 Center Road
Perry, OH 44081-0097

SUBJECT: PERRY NUCLEAR POWER PLANT, UNIT NO. 1 - ISSUANCE OF AMENDMENT CONCERNING FULL IMPLEMENTATION OF ALTERNATIVE SOURCE TERM (TAC NO. MF3197)(L-13-306)

Dear Mr. Harkness:

The U.S. Nuclear Regulatory Commission (NRC or Commission) has issued the enclosed Amendment No. 166 to Facility Operating License No. NPF-58 for the Perry Nuclear Power Plant (Perry), Unit No. 1. This amendment to the technical specifications (TS) and the associated revision to the Updated Safety Analyses Report is in response to your application of December 6, 2013, as supplemented by letters dated February 27, July 22, October 8, 2014, and February 4, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML13343A013, ML14059A221, ML14203A625, ML14282A218 and ML15042A214). More specifically, the amendment approves the use of updated accident source term employing an alternative source term (AST) methodology for the applicable design basis radiological analysis, and revises the TS definition of DOSE EQUIVALENT IODINE-131.

This licensing action is considered a full implementation of the AST. With this approval, the previous accident source term in the Perry design basis is superseded. The previous offsite and control room accident dose criteria expressed in terms of whole body, thyroid, and skin doses are superseded by the total effective dose equivalent criteria of Title 10 of the *Code of Federal Regulations* Part 50.67, or fractions thereof, as defined in Regulatory Guide 1.183.

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

Sincerely,

/RA/

Eva A. Brown, Senior Project Manager
Plant Licensing III-2 and
Planning and Analysis Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-440

Enclosures:
1. Amendment No. 166 to NPF-58
2. Safety Evaluation

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

FIRSTENERGY NUCLEAR OPERATING COMPANY

FIRSTENERGY NUCLEAR GENERATION CORP.

OHIO EDISON COMPANY

DOCKET NO. 50-440

PERRY NUCLEAR POWER PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 166
License No. NPF-58

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

   A. The application for amendment filed by FirstEnergy Nuclear Operating Company, et al., (the licensee, FENOC) dated December 6, 2013, as supplemented by letters dated February 27, July 22, October 8, 2014, and February 4, 2015, 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, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-58 is hereby amended to read as follows:

Enclosure 1

(2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 166 are hereby incorporated into this license. FENOC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. Accordingly, by Amendment No. 166, the license is amended to authorize revision to the Updated Safety Analysis Report (USAR), as set forth in the application dated December 6, 2013, as supplemented. The licensee shall update the USAR to incorporate the changes as described in the licensee's application dated December 6, 2013, as supplemented 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 USAR.

4. This license amendment is effective as of its date of its issuance and shall be implemented within 30 days of the date of issuance. The USAR changes shall be implemented in the next periodic update to the USAR in accordance with 10 CFR 50.71(e).

FOR THE NUCLEAR REGULATORY COMMISSION

Travis L. Tate, Chief
Plant Licensing III-2 and
Planning and Analysis Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical
Specifications and Facility Operating License

Date of Issuance: March 30, 2015

ATTACHMENT TO LICENSE AMENDMENT NO. 166

FACILITY OPERATING LICENSE NO. NPF-58

DOCKET NO. 50-440

Replace the following pages of the Facility Operating License and Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
License NPF-58 Page 4	License NPF-58 Page 4
TSs 1.0-2	TSs 1.0-2

renewal. Such sale and leaseback transactions are subject to the representations and conditions set forth in the above mentioned application of January 23, 1987, as supplemented on March 3, 1987, as well as the letter of the Director of the Office of Nuclear Reactor Regulation dated March 16, 1987, consenting to such transactions. Specifically, a lessor and anyone else who may acquire an interest under these transactions are prohibited from exercising directly or indirectly any control over the licenses of PNPP Unit 1. For purposes of this condition the limitations of 10 CFR 50.81, as now in effect and as may be subsequently amended, are fully applicable to the lessor and any successor in interest to that lessor as long as the license for PNPP Unit 1 remains in effect; these financial transactions shall have no effect on the license for the Perry Nuclear facility throughout the term of the license.

(b) Further, the licensees are also required to notify the NRC in writing prior to any change in: (i) the terms or conditions of any lease agreements executed as part of these transactions; (ii) the PNPP Operating Agreement; (iii) the existing property insurance coverage for PNPP Unit 1; and (iv) any action by a lessor or others that may have an adverse effect on the safe operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now and hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

FENOC is authorized to operate the facility at reactor core power levels not in excess of 3758 megawatts thermal (100% power) in accordance with the conditions specified herein.

(2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 166, are hereby incorporated into the license. FENOC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Antitrust Conditions

a. FirstEnergy Nuclear Generation Corp. and Ohio Edison Company

1.1 Definitions (continued)

CHANNEL FUNCTIONAL TEST	A CHANNEL FUNCTIONAL TEST shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY, including required alarm, interlock, display, and trip functions, and channel failure trips. The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps so that the entire channel is tested.
CORE ALTERATION	CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components within the reactor vessel with the vessel head removed and fuel in the vessel. The following exceptions are not considered to be CORE ALTERATIONS: a. Movement of source range monitors, local power range monitors, intermediate range monitors, traversing incore probes, or special movable detectors (including undervessel replacement); and b. Control rod movement, provided there are no fuel assemblies in the associated core cell. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.
CORE OPERATING LIMITS REPORT (COLR)	The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Plant operation within these limits is addressed in individual Specifications.
DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Federal Guidance Report (FGR) 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1989.

(continued)

UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 166 TO FACILITY OPERATING LICENSE NO. NPF-58

FIRSTENERGY NUCLEAR OPERATING COMPANY

OHIO EDISON COMPANY

PERRY NUCLEAR POWER PLANT, UNIT NO. 1

DOCKET NO. 50-440

1.0 INTRODUCTION

By application to the U.S. Nuclear Regulatory Commission (NRC, the Commission) dated December 6, 2013, as supplemented by letters dated February 27, July 22, October 8, 2014, and February 4, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML13343A013, ML14059A221, ML14203A625, ML14282A218 and ML15042A214) FirstEnergy Nuclear Operating Company (the licensee) requested changes to the technical specifications (TSs) and the updated safety analysis report (USAR) for the Perry Nuclear Power Plant (Perry), Unit No. 1. More specifically, the proposed amendment requests the use of updated radiological accident source term employing an alternative source term (AST) methodology for the applicable design basis accident (DBA) analyses, and revises the TS definition of DOSE EQUIVALENT IODINE (DEI)-131.

The February 27, July 22, October 8, 2014, and February 4, 2015, supplements, contained clarifying information and did not change the NRC staff's initial proposed finding of no significant hazards consideration dated April 15, 2014 (79 FR 21298).

2.0 REGULATORY EVALUATION

2.1 Background on 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 (ESFs). 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," dated March 23, 1962, as the basis DBA source terms. The power reactor siting regulation requires that a fission product release into containment be postulated and that offsite radiological consequences be evaluated against the guideline dose values given in Section 100.11 to Title 10 to the *Code of Federal Regulations* (10 CFR), "Determination of Exclusion Area, Low Population Zone, and Population Center Distance," which provides the guideline offsite dose limits in terms of whole body and thyroid dose and makes reference to TID-14844.

Enclosure 2

In December 1999, the NRC issued a new regulation 10 CFR, Section 50.67, "Accident Source Term," 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 two hours as opposed to the TID source term which assumed the release of the entire source term occurs instantaneously. In addition, in the revised source term, 95 percent of the radioiodine is assumed to be released as an aerosol, with the remaining 5 percent as a combination of inorganic and organic vapors. This is in contrast to the original TID source term which prescribed the opposite ratio, 95 percent of the iodine as vapor and 5 percent as aerosol.

Guidance for the implementation of the AST methodology is provided in Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." As part of the implementation of the AST, the total effective dose equivalent (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 design basis loss-of-coolant accident (LOCA) analysis is based upon a major postulated accident, or possible event, resulting in dose consequences not exceeded by those from any accident considered credible. Historically this accident analysis is referred to as the maximum hypothetical accident (MHA). It should be noted that the requirements of 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," ensure that the emergency core cooling system (ECCS) will prevent significant core damage during a design basis LOCA. Notwithstanding, the requirements of 10 CFR 50.46, the MHA deterministically assumes a substantial core melt with an appreciable release of fission products into the containment. Therefore, the MHA is a conservative surrogate to enable a deterministic evaluation of the response of a facility's ESFs such as the containment system. While the LOCA is typically the maximum credible accident, NRC staff experience in reviewing license applications has indicated the need to consider the dose consequences from other accidents such as the fuel handling accident (FHA). All DBAs are performed in an intentionally conservative manner in order to compensate for known uncertainties in accident progression, airborne activity product transport, and atmospheric dispersion.

2.2 Regulatory Requirements

Section 50.49 of 10 CFR, "Environmental Qualification of Equipment," requires that the safety-related electrical equipment which are relied upon to remain functional during and following the design basis events be qualified for accident (harsh) environment. This provides assurance that the equipment needed in the event of an accident will perform its intended function.

Section 50.67 of 10 CFR provides an optional provision for licensees to revise the source term used in design basis radiological consequence analyses and sets a 5 rem (roentgen equivalent man) TEDE limit onsite and 25 rem TEDE limit offsite for the duration of the accident.

Section 100.11 provides criteria for evaluating the radiological aspects of the proposed site. A footnote to 10 CFR 100.11 states that the fission product release assumed in these evaluations

should be based upon a major accident involving substantial meltdown of the core with subsequent release of appreciable quantities of fission products.

General Design Criterion (GDC) 17 of Appendix A to 10 CFR Part 50, requires, in part, that an onsite electrical power system and an offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system shall be to provide sufficient capacity and capability to assure that: (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

GDC 19 of Appendix A to 10 CFR Part 50, specifies a control room design criterion of 5 rem TEDE for facilities using the AST under 10 CFR Section 50.67.

GDC 26 of Appendix A to 10 CFR Part 50, requires that each reactor have two independent reactivity control systems of a different design, while GDC 29 requires that the reactivity control system be capable of accomplishing its safety function in the event of anticipated operational occurrences.

2.3 Approved Guidance

RG 1.23 (previously Safety Guide), "Meteorological Monitoring Programs for Nuclear Power Plants," Revision 1, presents criteria for an acceptable onsite meteorological monitoring program and the resulting meteorological database that may be used in this Standard Review Plan (SRP) section as input to the atmospheric dispersion estimates.

RG 1.52, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post Accident Engineered Safety Feature Atmosphere Cleanup Systems in Light Water Cooled Nuclear Power Plants," present methods acceptable to the NRC staff for meeting control room occupancy protection requirements.

Regulatory Position 1.2 of RG 1.183, indicates that complete implementation of the AST would upgrade all existing radiological analyses and address all characteristics of the AST, namely: composition and magnitude of the radioactive material, its chemical and physical form, and the timing of its release. Although a complete reassessment of all facility radiological analyses would be desirable, the NRC staff determined that recalculation of all analyses would generally not be necessary. Full implementation revises the facility's licensing basis to specify the AST in place of the previous accident source term. This applies not only to the analyses performed in the application, which may only include a subset of the plant analyses, but also to all future analyses. As a minimum for AST implementation, the LOCA must be analyzed using the guidance in RG 1.183, Appendix A.

Section 1.3.1 of RG 1.183, tabulates example aspects of plant operation that are predicated, in part, by DBA radiological calculations that could be affected by analysis assumption changes related to the source term implementation, or to associated plant modifications. The habitability of emergency response facilities is also identified in Section 1.3.1.

NUREG-0696, "Functional Criteria for Emergency Response Facilities," indicates that the TSC have the same habitability as the CR.

As stated in Regulatory Position 5.2 of RG 1.183, DBAs addressed in the appendices of RG 1.183 were selected from accidents that may involve damage to irradiated fuel. The inclusion or exclusion of a particular DBA in RG 1.183 should not be interpreted as indicating that an analysis of that DBA is required or not required. Licensees should analyze the DBAs that are affected by the specific proposed applications of an AST.

Regulatory Guide 1.194, "Atmospheric Relative Concentrations for Control Room [CR] Radiological Habitability Assessments at Nuclear Power Plants," provides guidance on determining atmospheric relative concentration (χ/Q) values in support of design basis control room radiological habitability assessments at nuclear power plants. This guide describes methods acceptable to the NRC staff for determining χ/Q values that will be used in CR radiological habitability assessments performed in support of applications for licenses and license amendment requests (LARs).

Regulatory Guide 1.196, "Control Room Habitability at Light-Water Nuclear Power Reactors, provides guidance on establishing the licensing bases for the control room envelop (CRE), its associated ventilation systems, and those located in, traversing or serving areas adjacent to the CRE.

NUREG-0933 Issue 187, "The Potential Impact of Postulated Cesium Concentration on Equipment Qualification" (ADAMS Accession No. ML012190402), indicates that for equipment exposed to the containment atmosphere, the TID-14844 source term and the gap and in-vessel releases in the AST produced similar integrated doses, and for equipment exposed to suppression pool water, the integrated doses calculated with the AST remain enveloped by those calculated with TID-14844 for the first 145 days post-accident for a BWR (boiling-water reactor, including the 30 percent vs. 1-percent release of cesium. It was concluded that there was no clear basis for back fitting the requirement to modify the design basis for equipment qualification to adopt the AST. There would be no discernible risk reduction associated with such a requirement.

Regulatory Position 6, and Appendix I of RG 1.189 addresses the requirements for assessing the impact of the difference in source term characteristics on environmental qualification (EQ) doses.

NUREG-0654/FEMA-REP-1, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," provides an acceptable method for implementing the agency's emergency planning regulations.

An acceptable implementation of an AST should demonstrate compliance with plant-specific licensing commitments made in response to the NUREG-0737, "Clarification of TMI Action Plan Requirements." Specific provisions of interest include the following:

- NUREG-0737 II.B.2, Post-accident Access Shielding, as it relates to post-accident radiation exposure incurred while performing necessary plant operations outside of the CR.
- NUREG-0737 II.B.3, Post-accident Sampling Capability, as it relates to post-accident radiation exposure during sampling operations.
- NUREG-0737 II.F.1, Additional Accident-Monitoring Equipment, as it relates to the ability of the monitors to operate during and following an accident and perform the intended function in the accident environment.
- NUREG-0737 III.D.1.1, Leakage Control, as it relates to post-accident radiation exposure.
- NUREG-0737 III.A.1.2, Supplement 1, Emergency Response Facilities, as it relates to maintaining emergency facilities in a safe, habitable condition under accident conditions by providing adequate protection against radiation and toxic gases.
- NUREG-0737 III.D.3.4, Control Room Habitability, as it relates to maintaining the control room in a safe, habitable condition under accident conditions by providing adequate protection against radiation and toxic gases.

NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR (light-water reactor] Edition":

- Section 2.3.3, "Onsite Meteorological Measurements Program," provides the acceptance criteria for the review of the onsite meteorological monitoring program and the resulting meteorological database that may be used in this SRP section as input to the atmospheric dispersion estimates
- Section 2.3.4, "Short-Term Atmospheric Dispersion Estimates for Accident Releases," provides the acceptance criteria for reviewing estimates of atmospheric dispersion factors at the exclusion area boundary (EAB), outer boundary of the low population zone (LPZ), and the CR for, among other things, postulated design-basis accident radioactive airborne releases
- Section 6.4, "Control Room Habitability System,"
- Section 6.5.2, "Containment Spray as a Fission Product Cleanup System," provides the acceptance criteria regarding the systems used to minimize iodine re-evolution as presented in the licensee's re-analysis of the radiological consequences for the LOCA.
- Section 13.3, "Emergency Planning"
- Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms

NUREG-0917, "Nuclear Regulatory Commission Staff Computer Programs for Use with Meteorological Data," provides methodologies for evaluating sequential hourly meteorological data used to characterize onsite meteorological conditions including, input to atmospheric dispersion modeling analyses.

NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants,"provides estimates of accident source terms that were more physically based and that could be applied to a BWR.

NUREG/CR-5950, "Iodine Evolution and pH control," December 1992.

NUREG/CR-5966, "A Simplified Model of Aerosol Removal by Containment Sprays," June 1993.

NUREG/CR-0009, "Technological Bases for Models of Spray Washout of Airborne Contaminants in Containment Vessels," October 1978.

NUREG/CR-6189, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments," is incorporated into the analysis code RADionuclide Transport and Removal And Dose Estimation (RADTRAD). Reduction in airborne radioactivity in the containment by natural deposition within the containment may be credited. The prior practice of deterministically assuming that a 50 percent plateout of iodine is released from the fuel is no longer acceptable to the NRC staff as it is inconsistent with the characteristics of the revised source terms.

NUREG/CR-6331, "Atmospheric Relative Concentrations in Building Wakes," Revision 1, May 1997, provides guidance for ARCON96 dispersion model used for confirmatory CR and onsite TSC χ/Q modeling runs.

3.0 TECHNICAL EVALUATION

3.1 Background

The licensee proposed a full implementation of the AST, in accordance with the guidance in RG 1.183, and Section 15.0.1 of the SRP. The scope of the licensee's AST analyses included the BWR DBAs identified in its USAR Chapter 15, "Accident Analysis," and described in RG 1.183 as events that could potentially result in significant CR and offsite doses. These DBAs include:

- LOCA,
- FHA,
- Control rod drop accident (CRDA), and the
- Main steamline break accident outside containment (MSLBOC).

By letter dated March 26, 1999 (ADAMS Accession No. ML021840462), the NRC revised TS 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)," and TS 3.6.1.9, "Main Steam Isolation Valve (MSIV) Leakage Control System (LCS)." The amendment reflected the implementation of the revised accident source term in NUREG-1465, and permitted the licensee to eliminate the MSIV Leakage Control System and increase the allowable leak rates of the MSIVs. This amendment served as the pilot plant application of the AST for a design basis LOCA. The licensee retained many aspects of the licensing basis from this initial AST application in the current AST LOCA evaluation.

By letter dated March 4, 2003 (ADAMS Accession No. ML023580025), the NRC approved the use of AST methodology for the FHA. The licensee did not propose changes to the FHA methodology in the current AST application, however, the licensee revised FHA dose calculations to support the use of the Global Nuclear Fuels (GNF)-2 fuel type.

The licensee completed a review of the impact of the proposed AST requirements applicability to the Perry NUREG-0737 commitments and modifications. The licensee determined that the AST implementation would not affect its current NUREG-0737 licensing basis. The Perry dose consequence analyses performed using the TID-14844 source term to show compliance with NUREG-0737 would be conservative relative to the AST primarily due to the instantaneous release assumption embedded in the TID-14844 source term. Therefore, the NRC staff finds that the licensee fully addressed the issue of maintaining consistency with the NUREG-0737 evaluations while incorporating the AST into the plant licensing basis for dose consequence analyses.

The DBA dose consequence analyses evaluated the integrated TEDE dose at the EAB for the worst 2-hour period following the onset of the accident. The integrated TEDE doses at the outer boundary of the LPZ and the integrated dose to a Perry CR operator were evaluated for the duration of the accident. The dose consequence analyses were performed by the licensee using the RADTRAD Version 3.03, computer code. The code estimates transport and removal of radionuclides and radiological consequence at selected receptors. The staff performs independent confirmatory dose evaluations, as needed, 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 this SE.

3.2 Proposed TS Change

The licensee proposed one change to TS Section 1.1 to revise the definition of DEI–131. The definition is revised to a DEI-131 value using dose conversion factors (DCFs) from the previously accepted EPA Federal Guidance Report (FGR) 11 "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," which is consistent with prior staff approvals and RG 1.183 guidance and is, therefore, acceptable to the NRC staff.

3.3 AST Analyses

3.3.1 Radiation Source Terms

The licensee analyzed the design basis events involving postulated fuel failures using a source term for GNF-2 fuel. RG 1.183, Regulatory Position 3.1, "Fission Product Inventory," states:

> 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 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 ORIGEN2 or ORIGEN-ARP.

The equilibrium core inventory source term used in the Perry AST analysis was performed by General Electric (GE) Hitachi Nuclear Energy, LLC (GEH) using plant-specific parameters as input into ORIGN01 P. ORIGN01 P is the GEH controlled version of the Oak Ridge National Laboratory code ORIGEN2. The ORIGEN01 P calculations assumed a core power equal to the current Perry licensed thermal power. In order to account for measurement uncertainty, the licensee increased the inventory provided by GEH by an additional 2 percent. Therefore, the dose consequence analyses are based on a reactor power level of 3833 megawatt thermal (MWt) based on 102 percent of the rated thermal power of 3758 MWt. For dose consequence evaluations that involve limited core damage such as the CRDA, Perry used a radial peaking factor of 2.0 to conservatively account for power distributions across the core.

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 megawatt days per metric ton of uranium (MWD/MTU) provided that the maximum linear heat generation rate does not exceed 6.3 kilowatt per foot (kw/ft) peak rod average power for burnups exceeding 54,000 MWD/MTU. The licensee's fuel burnup conforms to the restrictions described in RG 1.183. Therefore the NRC staff finds the source term used in the Perry AST dose consequence analyses is acceptable.

3.3.2 LOCA

3.3.2.1 LOCA Description

The licensee's LOCA analysis evaluated a double-ended guillotine break occurring in one of the four main steam lines upstream of the inboard MSIV, releasing reactor coolant to the drywell. When evaluating an AST LOCA for a BWR, it is assumed that the initial fission product release to the containment will last for 2 minutes and will consist of the radioactive materials dissolved or suspended in the RCS liquid. After 2 minutes, 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 1.5 hours. The licensee used the release fractions, timing characteristics, and radionuclide grouping as specified in RG 1.183, Table 1.

The licensee considered dose contributions from the following potential radioactive material release pathways:

- Primary containment bypass leakage directly to the environment;
- ESF system leakage; and
- MSIV leakage to the environment.

The licensee considered the following contributors to the CR habitability envelope (CRHE) analysis:

- Airborne activity inside the CR;

- Airborne cloud shine external to the CR;
- Containment shine to the CR; and
- Control Room Emergency Recirculation System (CRERS) filter shine.

In addition, the licensee evaluated the dose to the TSC using the same methodology incorporated into the CR dose analysis.

3.3.2.2 LOCA Source Term

The LOCA analysis assumes that the radioactivity released from the fuel is assumed to mix instantaneously and homogeneously throughout the free air volume of the drywell. The radioactivity release into the drywell is assumed to terminate at the end of the early in-vessel phase, which occurs at 2 hours after the onset of a LOCA.

The analysis credits control of pH in the suppression pool following a LOCA by means of injecting a pH buffering solution into the reactor core with the standby liquid control (SLC) system. The licensee completed an analysis that demonstrates the suppression pool pH will remain greater than 7.0 for the duration of the accident. The SLC system is safety-related and Seismic Category I.

In a letter dated March 26, 1999 (ADAMS Accession No. ML02180462), the NRC staff reviewed and approved the licensee's use of the SLC system to maintain the suppression pool pH greater than 7 following a LOCA. The injection of all of the sodium pentaborate from SLC provides sufficient buffering to maintain long-term pH. The NRC staff's review included independent verification of the licensee's calculations as well as laboratory titrations to confirm the calculation results. The licensee's analysis included strong acid generation from radiolysis of water and electric cables inside containment. As part of the current proposal, the licensee provided time-dependent pH curves for the duration of the post-accident scenario. The NRC staff verified that its previous review and approval of the SLC injection to maintain pH in the suppression pool remains valid. Based on its evaluation, the staff finds that the licensee's injection of SLC following an accident will maintain pH greater than 7 in the suppression pool.

The NRC staff found that the licensee has demonstrated the post-LOCA minimum suppression chamber water pH is greater than 7.0 for the 30-day duration of the accident. Therefore, the licensee concluded that the chemical form of radioiodine released into the containment is assumed to be 95 percent cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. Therefore, the NRC staff finds the use of SLC system to minimize the re-evolution of iodine into the containment atmosphere is acceptable.

3.3.2.3 Radioactive Material Transport in the Primary Containment

As described in Section 6.2 of the Perry Updated Safety Analysis Report, "Containment Systems," the containment and drywell have the capability to maintain functional integrity during and following peak transient pressures and temperatures that would occur following any postulated LOCA. In addition, the LOCA is postulated to occur simultaneously with a loss of offsite power (LOOP) and a safe-shutdown earthquake (SSE).

3.3.2.3.1 Natural Deposition in the Drywell and Containment

The licensee credited the reduction in airborne aerosol radioactivity in the containment by natural deposition, using a simplified model derived by correlation of the results of Monte Carlo uncertainty analyses of detailed models of aerosol behavior in the containment under accident conditions. Perry conservatively used the 10th percentile Power's Aerosol Decontamination Model in RADTRAD to account for the reduction in airborne radioactivity in the containment by natural deposition. The licensee only considered aerosol removal in the drywell and in the unsprayed region of the containment. This is because the containment spray will reduce the particle size distribution in the sprayed region of the containment significantly reducing the effect of natural aerosol deposition.

The licensee credited elemental iodine removal in the drywell and containment volumes using the methodology described in NUREG/CR-0009. The process of airborne elemental iodine removal by deposition to the walls in the drywell and containment is driven by the temperature differences between the surfaces and the atmosphere. The licensee calculated removal rate constants for the drywell and the containment based on their respective volumes and surface areas applying the bounding mass transfer coefficient from NUREG/CR-0009.

The NRC staff's review confirmed that the licensee applied approved assumptions and methodologies consistent with RG1.183. Therefore, the NRC staff finds that the licensee's approach to the evaluation of natural deposition in the drywell and containment has been demonstrated to be effective for fission product removal and retention.

3.3.2.3.2 Containment Spray Assumptions

The licensee's model for the removal of iodine was used to account for the reduction in airborne radioactivity in the containment due to the operation of the containment spray system. For Perry, the volume of the unsprayed containment region is 684,228 ft^3 while the volume of the sprayed containment region is 481,174 ft^3. Since the sprayed region of the Perry containment does not represent at least 90 percent of the total containment volume, the Perry containment building atmosphere would not be considered to be a single, well-mixed, volume. As a result, the licensee evaluated the mixing rate between the sprayed and unsprayed regions of the containment.

The licensee maintained the current licensing basis (CLB) assumption of a mixing rate between the unsprayed containment and the sprayed containment of 71,400 cubic feet per minute (cfm), as previously approved by the NRC staff in the SE dated March 26, 1999. In that SE, the NRC staff accepted the proposed mixing rate of 71,400 cfm on the basis of its review of the licensee's calculation that demonstrated that an adequate mixing flow existed between unsprayed and sprayed regions by natural convection.

The licensee applied limits on decontamination by limiting the elemental iodine removal by sprays to a decontamination factor (DF) of 200, and reduced the rate of particulate iodine removal by a factor of 10 once a DF of 50 was attained.

The NRC staff's review confirmed that the licensee applied approved assumptions and methodologies consistent with RG 1.183.. As the licensee's treatment of containment spray assumptions conforms to regulatory guidance, the NRC finds this approach acceptable.

3.3.2.3.3 Containment Leakage Pathway

The NRC staff reviewed the licensee's submittals to ensure that the leakage from the drywell into the primary containment is based on the steaming rate of the heated reactor core, with no credit for core debris relocation. The licensee assumed the CLB value of 3,000 cfm for the leakage from the drywell into the primary containment during the 2-hour period between the initial blowdown and termination of the fuel radioactivity release based on the steaming rate of the heated reactor core. The 3,000 cfm value was previously approved by the NRC staff in the LOCA calculation in the SE dated March 26, 1999, which states:

> The 3,000 cfm drywell bypass leakage rate is based upon large-break LOCA analyses performed with MELCOR on a Grand Gulf type [BWR Mark III] model (Reference 5). These analyses showed no relocation below the core plate, water level below the core plate, and an average steaming rate of approximately 2,800 cfm prior to quenching of the core at approximately 0.5 hours. Also, alternative water sources, such as the standby liquid control system, would not be available during station blackout sequences which comprise 96 percent of the core damage frequency for Grand Gulf according to NUREG-1150. Therefore, the staff concludes the use of 3,000 cfm for the drywell bypass leakage prior to 2 hours is reasonable for Perry.

RG 1.183, Appendix A, Regulatory Position 3.7, states:

> The primary containment (i.e., drywell for Mark I and II containment designs) should be assumed to leak at the peak pressure technical specification leak rate [L_a] for the first 24 hours. For BWRs, leakage may be reduced after the first 24 hours, if supported by plant configuration and analyses, to a value not less than 50 percent of the technical specification leak rate.

The licensee assumed that the primary containment leaks at the peak pressure TS leak rate, for the first 24 hours. After 24 hours, the licensee reduced the leakage rate to 0.69 L_a based on plant configurations and analysis. As the licensee's treatment of containment leakage conforms to regulatory guidance in RG1.183, the NRC finds this approach acceptable.

3.3.2.3.4 Containment Purge Pathway

The NRC staff reviewed the purge system to determine how the licensee considered the importance of minimizing the release of containment atmosphere to the environs following a postulated LOCA. The licensee stated that the Perry containment is not routinely purged during power operations. However, the licensee has retained a conservative containment purge dose consequence analysis based on the TID-14844 assumption of an instantaneous release of a substantial core melt source term into the containment atmosphere. This analysis is described

in USAR, Section 6.2.4.2.3, "Consideration of NRC Branch Technical Position CSB 6-4, 'Containment Purging during Normal Plant Operations."

This analysis is very conservative since the containment isolation valves are designed to close long before the onset of core damage given the assumptions incorporated into the AST. Even using the conservative containment purge assumption of an instantaneous substantial fuel melt, the calculated dose is less than 0.2 rem TEDE and would not have a significant impact on the licensee's LOCA dose analysis.

3.3.2.3.5 Containment Leakage to the Shield Building Annulus (Secondary Containment)

Regulatory Position 4.1 of RG 1.183, Appendix A, states, "Leakage from the primary containment should be considered to be collected, processed by ESF filters, if any, and released to the environment via the secondary containment exhaust system during periods in which the secondary containment has a negative pressure as defined in TSs. Credit for an elevated release should be assumed only if the point of physical release is more than two and one-half times the height of any adjacent structure."

RG 1.183, Appendix A, Regulatory Position 4.2, states, "Leakage from the primary containment is assumed to be released directly to the environment as a ground-level release during any period in which the secondary containment does not have a negative pressure as defined in technical specifications."

RG 1.183, Appendix A, Regulatory Position 4.3, states, in part, that, "The effect of high wind speeds on the ability of the secondary containment to maintain a negative pressure should be evaluated on an individual case basis."

During normal operation, the Perry shield building annulus is maintained at a slight negative pressure by operation of the ESF annulus exhaust gas treatment (AEGT) system. The licensee stated that although the annulus is expected to remain negative following a DBA, the dose analysis conservatively assumes that, for a short time period, containment pressure may not be maintained below the design negative pressure value of 0.25 inches water gauge. To account for this possibility, the licensee assumed that primary containment leakage is released directly to the environment for the first 40 seconds following the LOCA.

In addition, to ensure that the shield building annulus meets the design negative pressure value of 0.25 inches water gauge under adverse meteorological conditions, TS surveillance requirement (SR) 3.6.4.1.1, requires that at a frequency not to exceed 24 hours, annulus pressure be verified to be greater than or equal to 0.66 inches of vacuum water gauge.

The licensee conservatively modeled all releases including AEGT system releases as ground level releases. The 2,000 cfm AEGT system draws a negative pressure on the annulus and contains a high efficiency particulate air (HEPA) filter that is credited with 99 percent removal efficiency. The AEGT system also contains charcoal adsorbers; however, the licensee conservatively did not assign any removal credit to the charcoal adsorbers in the AEGT system. The licensee included a dose contribution from secondary containment bypass and conservatively assumed a bypass leak rate twice the maximum allowable value of 0.0504 L_a specified in TS SR 3.6.1.3.9. The licensee's analysis of containment leakage to the shield

building annulus conforms to the applicable regulatory guidance in RG 1.183 and is, therefore, acceptable to the NRC staff.

3.3.2.4 Assumptions on ESF System Leakage

Leakage from the ESF system develops when portions of the ECCS systems, located outside primary and secondary containments, circulate post-accident suppression pool water and leaks develop through packing glands, pump shaft seals and flanged connections. To evaluate the radiological consequences of ESF leakage, Perry used the deterministic approach as described in RG 1.183. This approach assumes that, except for the noble gases that are not soluble, all of the fission products released from the fuel mix instantaneously and homogeneously in the suppression pool water. Except for iodine, all of the radioactive materials in the suppression pool are assumed to be retained in the liquid phase. This source term assumption is conservative in that 100 percent of the radio-iodines released from the fuel are assumed to reside in both the containment atmosphere and in the suppression pool concurrently. The post-LOCA temperature of the liquid recirculated through the ESF systems is less than 212 degrees Fahrenheit. Therefore the licensee assumed that 10 percent of the iodine activity in the leaked ESF liquid becomes airborne in accordance with RG 1.183, Appendix A, Regulatory Position 5.5.

As specified in RG 1.183, Appendix A, Regulatory Position 5.2, the licensee assumed a value of 15.0 gallons per hour (gph) for ESF leakage, which is two times the administrative controls that limit ESF leakage. The licensee assumed that ESF leakage would begin at the onset of the gap release phase and continue for the 30-day duration of the accident evaluation. In addition to the evaluation of ESF leakage following the guidance in RG 1.183 as described above, the licensee also included the evaluation of the dose consequence of a postulated gross failure of a passive ESF component. This additional dose contribution was modelled as a 50 gallon per minute (gpm) leak starting 24 hours into the accident and lasting for 30 minutes.

The radiological consequences from the postulated ESF leakage are analyzed and combined with the consequences from other fission product release paths to determine the total calculated radiological consequences. The NRC staff finds that the assumptions and methodology used by the licensee to evaluate the dose consequence from post-LOCA ESF leakage is in accordance with the applicable regulatory guidance in RG1.183 and is, therefore, acceptable.

3.3.2.5 Assumptions on MSIV Leakage

The main steam lines (MSLs) in BWR plants, including Perry, contain MSIVs. Perry has four MSLs each of which is equipped with an inboard MSIV, an outboard MSIV, and a third isolation shutoff valve. These valves isolate the reactor coolant system in the event of a break in a steam line outside the primary containment, a design basis LOCA, or other events requiring containment isolation. These MSIVs along with the MSLs, up to and including the third isolation valve, are designed as Seismic Category I. Although the MSIVs are designed to provide a leak-tight barrier, it is recognized that some leakage through the valves will occur. Since the MSIVs are functionally part of the primary containment boundary, leakage through these valves provides a path for fission products that bypass the secondary containment. The initiating event in the Perry LOCA analysis assumes a double guillotine pipe rupture in one of the 4 MSLs upstream of the inboard MSIV. In addition, the analysis assumes the failure of all four third

main steam isolation shutoff valves to close as a result of a common power failure satisfying the single-failure criterion. The licensee conservatively assumed that the fission product leakage from the MSLs is released directly into the environment and that the leakage is assumed to begin immediately after the accident. RG 1.183, Appendix A, Regulatory Position 6, provides guidance for the evaluation of the radiological consequences from MSIV leakage that should be combined with other fission product pathways to determine the total calculated radiological consequences from a design basis LOCA.

Following the guidance in RG 1.183, Appendix A, Regulatory Position 6.1, the licensee assumed that the activity available for release via MSIV leakage is that activity determined to be in the drywell for evaluating containment leakage. Perry did not credit activity reduction by the steam separators and steam dryers or by iodine partitioning in the reactor vessel.

Following the guidance in RG 1.183, Appendix A, Regulatory Position 6.2, the licensee evaluated MSIV leakage at the maximum leak rate above in which the TS would require the MSIVs to be declared inoperable. Therefore, the licensee assumed a total MSL leak rate of 250 standard cubic feet per hour (scfh), which is the maximum allowable per TS, SR 3.6.1.3.10. The licensee assumed that the total MSIV leakage consists of: (1) 100 scfh through the steam line that is postulated to have broken, (2) 100 scfh through a second intact steam line, and (3) the remaining 50 scfh through a third intact steam line. No leakage is assigned to the remaining fourth intact steam line. Since the driver for MSIV leakage is containment pressure, after 24 hours the licensee reduced the assumed MSIV leakage by the same 69 percent reduction assumed for containment leakage.

RG 1.183, Appendix A, Regulatory Position 6.3, states:

> Reduction of the amount of released radioactivity by deposition and plateout on steam system piping upstream of the outboard MSIVs may be credited, but the amount of reduction in concentration allowed will be evaluated on an individual case basis. Generally, the model should be based on the assumption of well-mixed volumes, but other models such as slug flow may be used if justified.

The licensee maintained the assumptions for MSL deposition that were previously approved by the NRC staff in the SE dated March 26, 1999. The previously approved values for Perry MSL removal fractions are shown in Table 4 of this SE. The licensee assumed that all MSIV leakage past the outboard MSIV is released directly to the environment as a ground level release.

Appendix A of RG 1.183, Regulatory Position 6.5, allows radioactive material dose reduction from MSIV leakage due to holdup and deposition in MSL piping downstream of the MSIVs and in the main condenser, including the treatment of air ejector effluent by off-gas systems, if the components and piping systems used in the release path are capable of performing their safety function during and following an SSE. The licensee did not credit the hold up and deposition of MSIV leakage in the main condenser. The licensee did credit the safety-related seismic Category I piping downstream of the outboard MSIVs out to the third isolation valve in each MSL.

The licensee's analysis of the LOCA dose contribution due to MSIV leakage is consistent with the methodology previously approved in the SE dated March 26, 1999, and, therefore, remains acceptable to the NRC staff.

3.3.2.6 LOCA Control Room and TSC Habitability

Upon receipt of an ESF actuation system signal or a high radiation signal, the Perry CR heating, ventilation, and air conditioning (HVAC) system is designed to automatically isolate and activate the CRERS. For conservatism, the licensee assumed that the normal HVAC system continues to operate with an outside air intake of 6000 cfm and exhaust to the environment of 4800 cfm for the first 30 minutes of the accident analysis. After the first 30 minutes, the CRERS is assumed to be actuated by manual action. After actuation of the CRERS, there is no engineered intake of outside air and the CR pressure would eventually reach equilibrium with the surrounding areas. The licensee modelled the LOCA CR dose assuming the maximum CR unfiltered inleakage of 1375 cfm after actuation of the CRERS. The leakage out of the CR envelope is also modeled as 1375 cfm to avoid pressurization of the CR envelope.

The CRERS design recirculation flow rate of 30,000 cfm was conservatively modelled as a flow rate of 27,000 cfm in the dose consequence analysis. Each of the redundant CRERS filtration units contains an initial HEPA filter, charcoal adsorbers, and a final HEPA filter. Operation of the CRERS fans, charcoal adsorbers, and HEPA filters are credited in the LOCA dose analysis. The CRERS is an ESF system that is tested in accordance with RG 1.52. The current test acceptance criterion for the CRERS charcoal adsorbers requires a penetration of less than 2.5 percent. Following the guidance in RG 1.52, this testing requirement could justify using a charcoal adsorber removal efficiency of 95 percent in the dose analysis. For added operational margin, the licensee assumed an elemental iodine and organic iodide removal efficiency of 80 percent in the LOCA dose analysis.

In addition to the consideration of the dose resulting from airborne activity entering the CR, the licensee evaluated the contribution to the CR dose due to direct radiation shine from the containment building (0.13 rem), CRER filter shine (less than 0.008 rem) and the outside plume over the CR (0.002 rem). These dose evaluations were based on the TID-14844 source term and are therefore conservative relative to an AST analysis due to the TID-14844 assumption of an instantaneous core melt release.

The licensee evaluated the dose to TSC personnel in the same manner as the doses to the CR operators except for the TSC-specific atmospheric dispersion factors and other specific TSC data as shown in Tables 2-2 and 3 of this SE.

3.3.2. LOCA Analysis Conclusions

The licensee evaluated the radiological consequences resulting from the postulated LOCA and concluded that the radiological consequences at the EAB (exculsion area boundary) BOUNDARY), LPZ, and CR are within the dose guidelines provided in 10 CFR 50.67 and accident dose criteria specified in SRP Section 15.0.1. The NRC staff's review found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE. The assumptions found acceptable to the staff are presented in Table 4 and the licensee's calculated dose results are given in Table 1 of this SE.

The NRC staff finds that the EAB, LPZ, and CR doses calculated by the licensee for the Perry LOCA analysis meet the applicable regulatory criteria and are, therefore, acceptable.

3.3.3 CRDA

3.3.3.1 CRDA Description

The postulated Perry CRDA involves the rapid removal (i.e., drop) of a highest worth control rod resulting in a reactivity excursion. The postulated DBA event evaluated assumes that a control rod has been fully inserted and becomes stuck in this position. The control rod drive is assumed to be uncoupled and withdrawn. The rod subsequently becomes free and rapidly falls out of the core. The amount of positive reactivity introduced into the reactor core is at a rate consistent with the maximum control rod drop velocity, resulting in the insertion of positive reactivity and a localized power excursion. The licensee assumed in its evaluation of the CRDA, that as a result of the accident, fuel damage would occur consisting of localized damage to the fuel cladding with a limited amount of fuel melt occurring in the damaged rods. Consistent with RG 1.183, since the licensee has conservatively determined that fuel damage is postulated for this event, the impact of the normal coolant source terms is not evaluated in the dose consequence analysis.

3.3.3.2.1 CRDA Source Term

RG 1.183, Appendix C, Regulatory Position 1, states:

> Assumptions acceptable to the NRC staff regarding core inventory are provided in Regulatory Position 3 of this guide. For the rod drop accident, the release from the breached fuel is based on the estimate of the number of fuel rods breached and the assumption that 10 percent of the core inventory of the noble gases and iodines is in the fuel gap. The release attributed to fuel melting is based on the fraction of the fuel that reaches or exceeds the initiation temperature for fuel melting and on the assumption that 100 percent of the noble gases and 50 percent of the iodines contained in that fraction are released to the reactor coolant.

The licensee determined that the CRDA core source term are those associated with a DBA power level of 3833 MWt as discussed in Section 3.3.1 of this SE. The licensee included a radial peaking factor of 2.0 to account for power distributions across the core. The Perry core consists of 748 fuel bundles. Each GNF-2 fuel bundle contains 85.6 effective full-length fuel rods. For the CRDA, when the licensee performed a bounding fuel enthalpy analysis for GNF-2 fuel, approximately 1,200 fuel rods were determined to reach a fuel enthalpy of 170 calories per gram, at which point they can be assumed to experience cladding failure. The source term the licensee applied in the CRDA dose consequence analysis is conservatively based on the clad failure of 1,376 individual fuel rods contained in 16 fuel bundles. These 16 fuel bundles represent the 4 fuel bundle cell adjacent to the dropped control blade and one additional row surrounding the 4-bundle cell.

Consistent with RG 1.183 guidance, the licensee assumed that 10 percent of the core inventory of noble gases and halogens reside in the fuel gap and would be available for release from the

fuel rods experiencing fuel clad damage. The licensee determined that the maximum mass fraction in the damaged fuel that reaches temperatures in excess of the melting temperature is 0.0077. Consistent with RG 1.183 guidance, the licensee assumed the release of 100 percent of the noble gases and 50 percent of the iodines contained in that fraction of the fuel that is postulated to reach melt conditions. Consistent with the guidance in RG 1.183, Appendix C, Regulatory Positions 3.1 and 3.2, the licensee assumed that the gap activity and the activity from fuel pellet melting mixes instantaneously in the reactor coolant within the reactor pressure vessel with no credit for partitioning or removal by the steam separators.

The NRC staff has determined that the licensee's methods and assumptions used to evaluate the CRDA source term are consistent with RG 1.183 methods and assumptions. Therefore, based on the fact that the licensee used the fractional release values specified in RG 1.183, Appendix C, Regulatory Position 1, to evaluate the CRDA source term, the NRC staff finds that the licensee's evaluation of the CRDA source term is acceptable.

3.3.3.3 CRDA Activity Transport

The licensee evaluated two cases affecting activity transport in the CRDA. The first and most limiting case assumes that a loss of offsite power (LOOP) occurs at the time of the accident. A LOOP would cause the turbine stop and control valves to close, scram the reactor, and trip the condenser offgas system or mechanical vacuum pumps. The second case does not assume a LOOP and credits the significant holdup of both halogens and nobles in the low-temperature Perry offgas system. As shown in Table 1, the dose consequence for the CRDA with off-site power is considerably less than with the assumption of a coincident LOOP.
RG 1.183, Appendix C, Regulatory Position 3.3, states:

> Of the activity released from the reactor coolant within the pressure vessel, 100 percent of the noble gases, 10 percent of the iodine, and 1 percent of the remaining radionuclides are assumed to reach the turbine and condensers.

Consistent with this Regulatory Position, the licensee assumed that of the activity released from the reactor coolant within the pressure vessel, 100 percent of the noble gases, 10 percent of the iodine, and 1 percent of the remaining radionuclides are assumed to reach the turbine and condensers.

RG 1.183, Appendix C, Regulatory Position 3.4, states:

> Of the activity that reaches the turbine and condenser, 100 percent of the noble gases, 10 percent of the iodine, and 1 percent of the particulate radionuclides are available for release to the environment. The turbine and condensers leak to the atmosphere as a ground-level release at a rate of 1 percent per day for a period of 24 hours, at which time the leakage is assumed to terminate. No credit should be assumed for dilution or holdup within the turbine building. Radioactive decay during holdup in the turbine and condenser may be assumed.

Consistent with this Regulatory Position, the licensee assumed that of the activity that reaches the turbine and condenser, 100 percent of the noble gases, 10 percent of the iodine, and 1 percent of the particulate radionuclides are available for release to the environment at a leak rate of 1 percent per day, for a period of 24 hours.

The NRC staff has determined that the licensee's methods and assumptions used to evaluate the CRDA activity transport are consistent with RG 1.183 methods and assumptions. therefore, based on the fact that the licensee followed the assumptions outlined in RG 1.183, Appendix C, Regulatory Position 3.4, concerning retention of released activity in the condenser, the NRC staff finds that the licensee's evaluation of the CRDA activity transport is acceptable.

3.3.3.4 CR Habitability for the CRDA

In order to fully explore CR habitability for the CRDA, the license evaluated two cases. The first case assumes that the normal HVAC system is operating with a flow of 6,000 cfm ± 10 percent and is not isolated for the duration of the accident. For this case, the maximum inflow to the CR of 6,600 cfm is assumed and the emergency recirculation mode is not initiated. The second CR response case credits CRERS manual actuation after 30 minutes. When the CRERS is in operation, the outside makeup air is isolated and the CR envelope is not pressurized relative to adjacent areas. The licensee determined that the CR dose for the first case that does not credit CRERS operation is slightly higher than the second case. The licensee only reported the limiting CR dose for the first case.

The NRC staff has determined that the licensee's methods and assumptions used to evaluate the CR habitability for the CRDA are consistent with RG 1.183 methods and assumptions. Therefore, based on the fact that the licensee has demonstrated that the CR habitability design criteria can be met for a design rod drop accident without taking credit for the safety-related CRERS, the NRC staff finds that the licensee's evaluation of CR habitability for the CRDA is acceptable.

3.3.3.5 CRDA Analysis Conclusions

The licensee evaluated the radiological consequences resulting from the postulated CRDA and concluded that the radiological consequences at the EAB, LPZ, and CR are within the dose guidelines provided in 10 CFR 50.67 and accident specific dose criteria specified in SRP 15.0.1. The NRC staff's review found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE. The assumptions found acceptable to the NRC staff are presented in Table 5 and the licensee's calculated dose results are given in Table 1 of this SE. The EAB, LPZ, and CR doses estimated by the licensee for the CRDA were found to meet the applicable accident dose criteria and are, therefore, acceptable.

3.3.4 MSLBOC Accident

3.3.4.1 Analysis Summary

Accidents that result in the release of radioactive materials outside the containment are the result of postulated breaches in the nuclear system process barrier. In order to evaluate the possible effects of this event the licensee postulated a complete circumferential break on one of

the main steam lines immediately downstream of the outermost MSIV outside of the primary containment. The MSLBOC accident is described in the USAR, Section 15.6.4," Steam System Piping Break outside Containment." Perry plant is designed to immediately detect such an occurrence, initiate isolation of the broken line and actuate the necessary protective features.

Normally, the reactor operator will maintain reactor vessel water inventory and, therefore, core cooling with the reactor core isolation cooling (RCIC) system. Without operator action, the RCIC would initiate automatically on low water level following isolation of the main steam supply system. The core would be covered throughout the accident and there would be no fuel damage. Without taking credit for the RCIC water makeup capability and assuming high pressure core spray system failure, the automatic depressurization system will automatically initiate on low water level to ensure termination of the accident without fuel damage. Therefore, reactor vessel water level remains above the top of active fuel throughout the accident sequence, adequate core cooling is maintained and no fuel failures are assumed in the postulated MSLBOC.

The licensee assumed an MSIV closure time of 6.0 seconds based on TS 3.6.1.3 which requires that the MSIVs close within 5 seconds with an allowance of 1.0 second to account for signal actuation. The break mass released includes the amount of steam in the steam line and connecting lines at the time of the break, plus the amount of steam that passes through the valves prior to closure in accordance with regulatory guidance. Appendix D of RG 1.183 identifies assumptions acceptable to the NRC staff for the radiological analysis of an MSLB.

3.3.4.2 MSLBOC Accident Source Term

Appendix D of RG 1.183, Regulatory Position 2, states that if no or minimal fuel damage is postulated for the limiting event, the released activity should be the maximum coolant activity allowed by TS for the equilibrium case and for the pre-accident iodine spike case. Therefore, consistent with RG 1.183, the licensee evaluated the MSLBOC based on the maximum equilibrium reactor coolant DEI I-131 concentration of 0.2 μCi/gm and, in a separate analysis, evaluated the MSLBOC assuming a pre-accident iodine spike DEI concentration of 4.0 μCi/gm as specified in TS 3.4.8.

The NRC staff has determined that the licensee's methods and assumptions used to evaluate the MSLBOC source term are consistent with RG 1.183 methods and assumptions. Therefore, based on the fact that the licensee followed the assumptions outlined in Appendix D of RG 1.183, Regulatory Position 2, regarding the use of iodine spiking cases to define the source term for the MSLBOC, the NRC staff finds that the licensee's evaluation of the MSLBOC source term is acceptable.

3.3.4.3 MSLBOC Activity Transport

The licensee analyzed the MSLBOC by assuming the release pathway for this event is from the failed steam line directly to the atmosphere. The fission product release is modeled as an instantaneous puff release. This applies to both the maximum equilibrium coolant activity case and the pre-accident iodine spike case. No credit is taken for filtration by the Annulus Exhaust

Gas Treatment System or for mixing, holdup, dilution or decay prior to release. The release is assumed to terminate upon closure of the MSIVs at 6.0 seconds into the event. No additional leakage other than that directly from the break is assumed for this event.

The NRC staff has determined that the licensee's methods and assumptions used to evaluate the MSLBOC activity transport are consistent with RG 1.183 methods and assumptions. Therefore, based on the fact that the licensee followed the assumptions outlined in Appendix D of RG 1.183, Regulatory Position 4, regarding the direct fission product release to the atmosphere, the NRC staff finds that the licensee's evaluation of the MSLBOC activity transport is acceptable.

3.3.4.4 CR Habitability for the MSLBOC Accident

For both the equilibrium case and the pre-accident iodine spike case the licensee analyzed two separate CR ventilation operating modes. The first CR ventilation mode assumes that the normal CR ventilation system is operating with a flow of 6,000 cfm and is not isolated for the duration of the accident. The licensee incorporated the test criterion upper bound flow rate of 6,600 cfm in the dose analysis and assumed and the emergency recirculation mode is not initiated.

For the second CR ventilation mode analyzed, the licensee assumed that the CR ventilation system operates in the emergency recirculation mode with outside makeup air isolated and the CR envelope not pressurized relative to adjacent areas. An unfiltered inleakage to the CR of 1,375 cfm is assumed to begin at time zero and is assumed for the remaining duration of the accident. In addition to this unfiltered inleakage, the licensee assumed an additional unidentified unfiltered inleakage of 10 cfm to account for ingress and egress making the total unfiltered inleakage into the CR for this case 1385 cfm. Emergency recirculation is assumed to begin at 30 minutes. The NRC staff notes that the licensee only reported the limiting CR dose for the first case, which does not credit CRERS operation.

The NRC staff has determined that the licensee's methods and assumptions used to evaluate the CR habitability for the CRDA are consistent with RG 1.183 methods and assumptions. Therefore, based on the fact that the licensee has demonstrated that the regulatory criteria can be met for a MSLBOC without taking credit for the safety related CRERS, the NRC staff finds that the licensee's evaluation of CR habitability for the MSLBOC is acceptable.

3.3.4.5 MSLBOC Accident Analysis Conclusions

The licensee evaluated the radiological consequences resulting from the postulated MSLBOC and concluded that the radiological consequences at the EAB, LPZ, and CR are within the dose guidelines provided in 10 CFR 50.67 and accident specific dose criteria specified in SRP, Section 15.0.1. The NRC staff's review found that Perry used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE. The assumptions found acceptable to the NRC staff are presented in Table 6 and the licensee's calculated dose results are given in Table 1 of this SE. The EAB, LPZ, and CR doses estimated by the licensee for the MSLBOC were found to meet the applicable accident dose criteria and are, therefore, acceptable.

3.4 Atmospheric Dispersion Estimates

3.4.1 Atmospheric Dispersion Factors

3.4.1.1 Control Room Atmospheric Dispersion Factors

3.4.1.1.1 Background

The licensee's request is for full implementation of the AST methodology, therefore, this methodology did not require the re-calculation of atmospheric dispersion factors (χ/Q values) at the CR. For the design-basis LOCA, the licensee used the Perry Control Room χ/Q values approved by the NRC in a letter dated March 26, 1999 (ADAMS Accession No. ML021840462). That NRC staff notes that the CR χ/Qs approved in 1999 were not determined following the traditional approach of atmospheric dispersion modeling, but rather based on the results of site-specific experimental data collected during an onsite tracer gas study conducted in September 1985. Based on information available to the NRC staff, that field measurement program was initiated in recognition of:

- The complicated air flow patterns that exist in a complex of industrial buildings of varying shapes and dimensions and the turbulent wakes that they cause;
- The locations of potential accident release points (sources) relative to the air intakes to the CR (receptors); and
- The prevailing state of the art (with its limitations, and conservatisms) of atmospheric dispersion modeling tools available at that time for evaluating these situations.

Evaluation of that field measurement program, its results, and accepted alternatives to the χ/Qs proposed by the licensee based on the confirmatory modeling analyses by the NRC staff at that time is summarized in Section 3.6 of the 1999 AST approval. Atmospheric dispersion factors based on the licensee's tracer study, the licensee's revised χ/Q values, and the NRC staff's confirmatory modeling results, are summarized below from the 1999 AST approval.

Accident Period	Licensee's Tracer Study χ/Qs	Licensee's Revised χ/Qs	Staff's Confirmatory (ARCON96) χ/Qs
0-8 hrs	7.0 E-5	3.5 E-4	1.8 E-3
8-24 hrs	5.6 E-5	2.1 E-4	7.3 E-4
24-96 hrs (1-4 days)	4.3 E-5	1.1 E-4	4.7 E-4
96-720 hrs (4-30 days)	1.5 E-5	5.75 E-5	2.9 E-4

3.4.1.1.2 USAR Errors

In the December 6, 2013, submittal, the licensee proposed to correct Table 15.6-13 of the USAR to reflect two errors to the accident-related χ/Q values at the CR. Specifically, the licensee proposed to correct the χ/Qs applicable to the 24- to 96-hour and the 96- to 720-hour time intervals consistent with the 1999 AST LOCA evaluation approval. The NRC staff

observed that the change to the 24- to 96-hour (1 to 4 day) χ/Q value represented an order of magnitude increase (i.e., from 1.1 E-5 to 1.1 E-4) and that the latter value appeared to represent a change in reporting precision (i.e., from 5.8 E-5 to 5.75 E-5). The NRC staff was concerned whether the erroneous values in the existing USAR reflected the credited values in the analysis used in determining the CR dose for any of the affected accident scenarios. In the July 22, 2014, supplement, the licensee clarified that the values in the current USAR were "not currently credited to determine control room dose in any affected scenario."

The NRC staff reviewed the complete set of accident-related χ/Qs (i.e., for the 0- to 8-hour, 8- to 24-hour, 24- to 96-hour, and 96- to 720-hour time intervals), including the corrected values, applied to the LOCA scenario provided in Attachment 1 to the July 22, 2014, supplement. The licensee states that these χ/Q values were utilized in the LOCA calculations for "both the current licensing basis (CLB) calculation and the proposed licensing basis calculation". The NRC staff notes that the 0- to 8-hour χ/Q value listed on Page 8 of Attachment 1 and in USAR Table 15.6-13 also applies to 0- to 2-hour accident time interval. The NRC staff confirmed that the proposed corrected values are consistent with the approved CR χ/Q values found in the March 26, 1999, AST evaluation approval. The licensee further indicated that those same χ/Q values are applied to the CR dose analyses associated with the CRDA and MSLBOC accident scenarios.

Based on confirmation that the proposed accident-related χ/Qs used for evaluating CR dose for the LOCA scenario are the same as those previously approved, the NRC finds it acceptable and reasonable for use in evaluating the dose consequences of the CRDA and MSLBOC accident scenarios for Perry.

3.4.1.2 TSC Atmospheric Dispersion Factors

In the October 8, 2014, supplement, the licensee indicated that Perry has a primary TSC, located onsite in the basement of the Service Building, and "an alternate offsite TSC located more than 10 miles away from the plant site" that is "co-located at a facility that also serves as the offsite emergency operations facility (EOF), 10.44 miles from the plant". The licensee stated that "[d]ue to the offsite TSC's distance from the site, determination of χ/Q values and calculations of radiological dose to occupants is not necessary, consistent with guidance in Supplement 1 to NUREG-0737." In the December 6, 2013, submittal, the licensee included the calculation of χ/Q values factors at the onsite TSC. These χ/Q values directly support the design basis radiological dose analyses. The NRC reviewed the past NRC approvals issued in 1999 and 2003 and identified that the methodology for establishing χ/Q values used in the dose calculation for the emergency planning facilities serving as the TSC had not been previously reviewed and approved. In the October 8, 2014, supplement, the licensee indicated that two of TSC LOCA dose studies had been performed in 1980 and 1988 by the licensee and some information was provided in support of the FHA approval using the AST, but confirmed that the NRC staff had not approved any dose assessments for the TSC. The NRC staff reviewed the licensee's conclusion against Table 1 in Supplement 1 to NUREG-0737 and concluded that as the onsite TSC is the credited emergency response facility, approval of the χ/Q values for the alternate TSC is not required.

Similar to the χ/Q values at the CR air intakes, the χ/Q values at the air intake to the onsite TSC were based on the results of the onsite tracer gas study conducted in September 1985, discussed in Section 3.4.1.1.1. However, because measurements during the tracer study were not made at the onsite TSC air intake, the TSC χ/Q values were scaled by the licensee from the CR χ/Qs established in the 1999 AST evaluation approval.

The licensee provided the resulting χ/Q values at the onsite TSC in the October 8, 2014, supplement. These factors are summarized below.

Accident Period	Licensee's Originally Proposed Scaled TSC χ/Qs
0-8 hrs	5.1 E-5
8-24 hrs	4.1 E-5
24-96 hrs (1-4 days)	3.1 E-5
96-720 hrs (4-30 days)	1.1 E-5

The NRC staff reviewed the tracer gas study results as discussed in the previous 1999 and 2003 NRC approvals and found that the potential uncertainties in the tracer study results were appropriately reflected in Section 15.6.5.5.1.10 of the current USAR. The NRC staff further notes that the onsite TSC χ/Q values, as proposed in the original LAR of December 6, 2013, did not account for the same adjustments to the tracer gas study-based CR χ/Qs reflected in Amendment 103. Nevertheless, the same uncertainties (at a minimum) should apply to the TSC χ/Q values since they were developed from the CR χ/Qs. In the December 6, 2013, submittal the licensee indicated:

> The calculated dose values are reasonable since, as noted previously in the fuel handling accident AST submittal, the TSC ventilation intake is farther away from the containment structure and from ventilation system release points than the control room intake, and the TSC intake is at a lower elevation by more than 60 feet. Since the dispersion of a plume at the TSC intake at a greater distance and lower elevation would be correspondingly better than at the control room intake, the evaluations concluded that the 5 rem TEDE limit would be met for the TSC as well.

The NRC staff found that those arguments would be reasonable for typical straight-line plume dispersion; however, one of the indicated purposes for having conducted the tracer gas study was in recognition of the complicated flow regime (including possible air stagnation) that exists in a complex of industrial buildings such as Perry. As a result, the NRC staff was concerned that the proposed, scaled X/Q values for the onsite TSC were not adjusted to account for the same measurement uncertainties in the tracer gas study performed for the design-basis CR X/Qs. In the February 4, 2015, supplement, the licensee reiterated that the onsite TSC study was "... considered a realistic rather than [a] design-basis study." The licensee went on to indicate:

> FENOC has adjusted the onsite TSC X/Q values in a similar fashion as was done for the control room X/Q values in 1998, to make them more conservative...[t]his results in more conservative X/Q values for the onsite TSC than those submitted previously...[and that] these more conservative values are considered adequate to account for the measurement uncertainties...

Attachment 2 to the February 4, 2015, supplement, explains the method by which the TSC χ/Q values were calculated, including:

- Identifying the original TSC χ/Q values using the Murphy-Campe modeling approach and the original 3-year licensing basis meteorological data set;

- Scaling the 0- to 8-hour time interval TSC χ/Q based on a reduction factor determined from modeled CR χ/Q values, again using the Murphy-Campe approach and ratios between χ/Qs using the original 3-year meteorological data set and a 7-year meteorological data set subsequently available;

- Further scaling of the 0- to 8-hour time interval TSC χ/Q value by applying both wind direction and wind speed variability adjustment factors, from the Murphy-Campe guidance, to estimate the χ/Qs for the longer accident time intervals (i.e., 8- to 24-hours, 24- to 96-hours, and 96- to 720-hours); and

- Scaling these χ/Q values for all accident time intervals using factors agreed upon between the licensee and the NRC staff in establishing the CLB Control Room χ/Q values as part of 1999 AST LOCA evaluation approval.

The resulting onsite TSC χ/Q values are summarized below.

Accident Period	Licensee's Revised Re-Scaled TSC χ/Qs
0-8 hrs	2.6 E-4
8-24 hrs	1.6 E-4
24-96 hrs (1-4 days)	8.1 E-5
96-720 hrs (4-30 days)	4.25 E-5

The NRC staff considers the approach discussed in the February 4, 2015, supplement to be reasonable and acceptable for addressing measurement uncertainties in the tracer gas study to support determination of the onsite TSC χ/Q values.

3.4.2 Confirmatory CR and TSC Atmospheric Dispersion Factors

As part of its review of December 6, 2013, submittal, and the supplements dated July 22, 2014, October 8, 2014, and February 4, 2015, the NRC staff performed confirmatory dispersion

modeling analyses. The purpose of these analyses was to independently estimate impacts at the onsite TSC air intake due to accident-related radiological releases in order to determine the reasonability of the χ/Q values proposed by the licensee in its submittals. These χ/Q values are a direct input to the dose analyses conducted in support of full implementation of the AST methodology.

The NRC staff recognizes that the CR and TSC χ/Q values proposed by the licensee are based on the results of an onsite tracer gas study conducted in September 1985, because of the complex dispersion that can occur in a complex of industrial buildings like the Perry site. There are also limitations in available dispersion modeling tools (e.g., sometimes overly conservative dispersion estimates). As a result, Regulatory Position 7 in RG 1.194 provides for the use of site-specific experimental data in design-basis radiological assessments. The NRC staff considers that the licensee's use of this previously reviewed and approved approach, is acceptable for estimating accident-related χ/Qs which directly support the dose consequence evaluations at the Perry CR and onsite TSC.

3.4.2.1 Meteorological Data

The site-specific meteorological measurements made in accordance with Safety Guide 23 (predecessor to RG 1.23) are used to characterize, in part, the Perry site including input to atmospheric dispersion modeling estimates at the EAB and the LPZ due to accident releases. The period of record (POR) for these measurements covers seven annual cycles including May 1, 1972, through April 30, 1974, and September 1, 1977, through August 31, 1982. The relationship of those χ/Q values to this review are discussed further in Section 3.4.3.

Meteorological measurements for the indicated 7-year POR were likewise not used by the licensee in estimating χ/Q values associated with CR habitability dose evaluations. Rather, short-term, onsite meteorological data were collected in conjunction with a site-specific tracer gas study aimed at characterizing the dispersion of hypothetical DBA releases within the reactor building complex, between potential points of release and the air intakes to the CRs of Unit 1 and then proposed Unit 2. The results of that field study, conducted in September 1985, were accounted for in the 1999 AST evaluation approval.

Subsection 3.4.1.2 of the December 6, 2013, submittal, discusses DBA LOCA dose estimates at the onsite TSC. The NRC staff's review of these estimates included confirmatory calculations using the ARCON96 dispersion model which requires sequential hourly meteorological data as one of its inputs. In its February 4, 2015, supplement, the licensee provided seven years of onsite meteorological data: as a continuous data file formatted according to Appendix A to RG 1.23 and as 7 separate annual data files formatted for input to the ARCON96 model. The POR covered by these data includes calendar years 1993 through 1999 (see Attachment 3 to the referenced RAI response). This POR is the same duration as the onsite meteorological data represented in the CLB and so is considered to be acceptable from that perspective.

The NRC staff reviewed the hourly meteorological data provided by the licensee from the standpoint of its long-term representativeness and in terms of its acceptability for use as input to the NRC staff's confirmatory dispersion modeling analyses. These checks were made in consideration of the meteorological monitoring guidance in RG 1.23 and NUREG-0917 as cited in SRP, Section 2.3.3, and the meteorological data input requirements for the ARCON96

dispersion model based on the guidance in NUREG/CR-6331. Quantitative and qualitative data validation checks included, among others; data recovery and completeness: date/time stamps; joint recovery and characteristics of wind speed, wind direction, and atmospheric stability; variation of atmospheric stability conditions by time of day and other weather elements; and identification and frequency of calm wind conditions.

Based on this series of checks, it is reasonable to conclude that the POR for the CLB is representative of long-term dispersion conditions in the Perry site area and that the meteorological data provided by the licensee, as adjusted, is acceptable for use in the NRC staff's confirmatory modeling analyses.

3.4.2.2 Atmospheric Dispersion Modeling

In the February 4, 2015, supplement, the licensee provided additional atmospheric dispersion model input information. The confirmatory modeling scope for evaluating impacts at the onsite TSC air intake included five modeling runs – two each for potential releases from the respective Unit 1 and proposed Unit 2 containments and vents, and one model run to evaluate the MSLBOC accident scenario at that receptor. As indicated previously, the NRC staff made an additional modeling run to estimate impacts at the Unit 1 CR air intake, due to the postulated MSLBOC accident scenario, to compare against the CLB results for the design-basis LOCA. In this case, only Unit 1 was evaluated because it is the only operational unit and it is located closest to the MSLBOC release point.

The licensee noted in Attachment 2 to the February 4, 2015, supplement, that the χ/Q values as originally modeled "represent the worst case of those comprised of the combination of Units 1 and 2 containments and vents with the Technical Support Center intake" and that "because Unit 2 is not operational, the values...are considered conservative in that they consider the Unit 2 containment which would have a greater impact on the X/Q due to it being closer to the TSC intake". For comparability, the confirmatory modeling scope considered Unit 2-related sources and receptors.

The table below lists ARCON96 model input parameters based on the information provided in the referenced February 4, 2015, RAI response and other available plan and section view drawings of the plant. The respective scenarios are identified by characters that identify the receptor (i.e., TSC=TSC air intake and U1CR=Unit 1 CR air intake) followed by letter indicators for the source (e.g., U1C=Unit 1 Containment, U2V=Unit 2 Vent, MSBOC or MSOC=Main Steam Line Break Outside Containment).

Scenario	Release Type	Release Height (m)	Building Area (m2)	Distance to Receptor (m)	Intake Height (m)	Elevation Difference (m)	Direction to Source (Deg)
TSC-U1C	Ground-Level	35.8	4055.4	103.0	1.8	0	33
TSC-U1V	Ground-Level	40.8	4055.4	84.2	1.8	0	40
TSC-U2C	Ground-	35.8	4055.4	78.9	1.8	0	64

	Level						
TSC-U2V	Ground-Level	40.8	4055.4	82.8	1.8	0	41
TSCMSBOC	Ground-Level	8.4	2627.8	148.1	1.8	0	21
U1CRMSOC	Ground-Level	8.4	5338.8	92.4	20.60	0	30

The χ/Q values for the onsite TSC as originally proposed and subsequently revised by the licensee (see Subsection 3.4.1.2) and the results of the NRC staff's confirmatory ARCON96 modeling analyses for impacts at the TSC (based on the input information discussed in Subsections 3.4.2.1 and 3.4.2.2) are summarized in the table below.

Accident Period	Licensee's Originally Proposed Scaled TSC χ/Qs	Licensee's Revised Re-Scaled TSC χ/Qs	Staff's Confirmatory (ARCON96) TSC Max Unit 2 χ/Qs	Staff's Confirmatory (ARCON96) TSC Max Unit 1 χ/Qs
0-2 hrs	5.1 E-5	2.6 E-4	8.18 E-4	6.72 E-4
0-8 hrs	5.1 E-5	2.6 E-4	6.36 E-4	4.93 E-4
8-24 hrs	4.1 E-5	1.6 E-4	2.61 E-4	2.03 E-4
24-96 hrs (1-4 days)	3.1 E-5	8.1 E-5	1.66 E-4	1.30 E-4
96-720 hrs (4-30 days)	1.1 E-5	4.25 E-5	1.15 E-4	9.65 E-5

The licensee's χ/Q values for the 0- to 2-hour and 0- to 8-hour accident time periods (as originally proposed and as revised) were initially developed from a 1-hour maximum χ/Q value based on the licensee's onsite tracer gas study. These χ/Q values apply to both time intervals. On the other hand, the NRC staff's confirmatory modeling results are estimated directly by the ARCON96 dispersion model for all accident time periods.

The NRC staff's confirmatory modeling results for the proposed Unit 2 and Unit 1 represent the maximum χ/Q values based on either the impact due to an associated release from the corresponding Containment Building or plant vent. In the case of Unit 2 (and for both units evaluated), the overall maximum χ/Qs represent a release from the Unit 2 containment to the TSC air intake, likely due to the fact that this is the closest source-receptor pair. The maximum χ/Qs listed for Unit 1 represent a release from the Unit 1 plant vent, the closest source-receptor pair for the Unit 1 releases evaluated.

Comparisons between the licensee's revised and re-scaled χ/Qs provided with the February 4, 2015, supplement, and either the NRC staff's confirmatory modeling results for the proposed Unit 2 or Unit 1 show that the NRC staff's values for all accident time intervals range from a factor of about 1.5 to 3.5 times higher. The NRC staff recognized and expected the confirmatory ARCON96-modeled χ/Qs to be conservatively higher compared to actual field measurements developed from the onsite tracer gas study. These ratios are significantly less

(about an order of magnitude) than the NRC staff-modeled to licensee-proposed χ/Q ratios in Section 3.6 of the SE accompanying the 1999 AST evaluation approval. These differences are likely due to modeling assumptions made in the NRC staff analyses and those associated with the 1999 AST evaluation approval.

As a result of the reviews and confirmatory analysis performed, the NRC staff determined that the licensee's revised and re-scaled χ/Q values for the onsite TSC are reasonable and acceptable for application to the design-basis dose evaluations for full implementation of the AST methodology at Perry. The staff also finds that the χ/Q values at the Unit 1 CR air intake, based on its confirmatory modeling analysis for the MSLBOC accident scenario, compare well with the CLB χ/Qs for the LOCA scenario. The NRC staff notes χ/Q values at the air intake to the onsite TSC for the MSLBOC accident scenario are less than the χ/Q values for the Unit 1 and proposed Unit 2 source-receptor pairs. Therefore, the NRC staff has concluded that the MSLBOC does not represent the controlling accident scenario.

3.4.3 EAB/LPZ Atmospheric Dispersion Factors

The main focus of the December 6, 2013, submittal, is on full implementation of the AST methodology. As such, this methodology did not involve the re-calculation of χ/Q values at the EAB and the outer boundary of the LPZ. Rather, for the design-basis LOCA, CRDA, and MSLBOC scenarios evaluated by the licensee, the χ/Q values associated with the design-basis LOCA as listed in Table 2.3-24 of the current USAR were assumed to apply to all three accident scenarios (see Table 6-3, Table 3-2, Table 6-1 in Addenda 4, 5, and 6, respectively, of the December 6, 2013, submittal). The NRC staff finds this approach to be acceptable with respect to the LOCA and CRDA scenarios because releases to the ambient air would be associated with leakage from the Containment Building and/or from the plant vent. Therefore, the same χ/Q values at the EAB and LPZ receptors would apply.

However, the NRC staff notes that Figure 1 of Addendum 6 to the December 6, 2013, submittal, shows that the release path to atmosphere for the MSLBOC accident scenario would occur from the Auxiliary Building. This is a slightly different location than that assumed for the LOCA and CRDA scenarios (i.e., from the outer edge of the Containment Building as indicated in Subsection 2.3.4.2 of the current USAR). Based on information in its February 4, 2015, supplement, the NRC staff assumed that the release point for the MSLBOC accident scenario is located about 20 m north (relative to Plant North) of the Containment Building.

For the current configuration at Perry (i.e., only Unit 1 operating), plume travel distances between the assumed release points for the LOCA and CRDA dispersion analyses and receptors on the existing EAB and LPZ would then be reduced for some direction sectors (i.e., typically those sectors with a northerly component). In turn, shorter plume travel distances could result in higher χ/Q values at these receptors. Consequently, the NRC staff was concerned that the distances to the EAB and the LPZ receptors from the potential MSLBOC release point could be less than the distances implied by assuming that the same χ/Q values for the DBA apply for the MSLBOC release scenario.

In the February 4, 2015, supplement, the licensee acknowledged:

- The reduction in distance would most likely occur north and west of the plant whose sectors consist of Lake Erie:

- Distances would be reduced by a small amount in north-northeast and northeast sectors which are over land;

- Dose for populated areas to the south of the plant would decrease due to the slightly increased distance to the current EAB and LPZ for the majority of the wind rose sectors; and

- The 2010 permanent resident population in the west through north-northeast sectors is zero out to 10 miles from the plant.

In addition, the licensee identified control authorities, summarized notification procedures and responsibilities, and expected responses by recreational boaters and mariners within the affected area on Lake Erie in the event of an emergency.

As part of its review, the NRC staff noted that the controlling 0- to 2-hour accident-related χ/Q value at the EAB occurred in the northwest sector (see Table 2.3-24 of the current USAR), further suggesting a possible need for confirmatory modeling to determine whether slightly shorter plume travel distances would result in higher χ/Q values compared to the CLB. This EAB receptor is located over water in Lake Erie as illustrated by Figure 2.1-4 of the current USAR. As such, the NRC staff considered neither the statement of zero population out to 10 miles from the plant nor the availability of emergency response procedures pertaining to over-water receptors under accident conditions as a basis for not including a confirmatory modeling analysis as part of its review scope.

The NRC staff considered the magnitude of the resulting doses at the EAB and LPZ for the MSLBOC accident scenarios, evaluated by the licensee and reported in Sections 11.1 and 11.2 of Addendum 6 to the initial LAR submittal, as the basis for not performing any confirmatory modeling runs. The NRC confirmed that the resulting doses are an order of magnitude or more below the corresponding dose limits and the slight decrease in distance to the EAB receptors due to a different release location is not expected to challenge that demonstration of compliance for this accident scenario. Based on resulting doses being below the corresponding dose limits, the NRC staff has concluded that the licensee's proposed application of the CLB χ/Q values for offsite dose evaluations at the EAB and LPZ distances is acceptable for the design-basis LOCA, CRDT, and MSLBOC scenarios for full implementation of the AST methodology at Perry.

3.5 Review of Habitability for the Emergency Response Facilities

On page 9 of the December 6, 2013, submittal, the licensee indicated that only the TSC has specific dose criterion for radiological habitability, which the licensee identified as 5 rem TEDE. The licensee performed dose studies of the TSC and provided a summary of the large break LOCA (LBLOCA). The licensee indicates that the LBLBOCA analyses were performed in accordance with RG 1.183, using the NRC sponsored RADTRAD code. The analyses estimate doses for the EAB, LPZ, CR, and TSC, with a projected dose of 0.7 rem TEDE.

The licensee stated that its EOF is located beyond 10 miles and that accordingly, there is no specific dose criterion. The staff finds this exclusion to be acceptable because Table 2 of NUREG-0696 provides no radiological habitability criteria for an EOF beyond 10 miles.

The licensee's initial submittal did not address the impact of the implementation of the AST and the proposed TS change on the analysis that determined the numeric thresholds for radiation monitor. There are two classes of radiation monitors involved with the emergency action levels (EALs): (1) radiological effluent monitors, and (2) area monitors used in the fission product barrier EALs. Such calculations are dependent on source term assumptions. The licensee explained the impact of this licensing action on these EAL thresholds, and indicated in the supplement dated July 22, 2014, that there was no impact. The licensee's conclusion was based upon the following:

a. The Perry EAL scheme was based upon NUMARC/NESP-007 "Methodology for Development of Emergency Action Levels." The radiation monitor EAL thresholds were based, in part, on a source term derived from the licensing basis LOCA analysis.

b. In the early 1990s, the licensee adopted a source term derived from NUREG-1228, "Source Term Estimation during Incident Response to Severe Nuclear Power Plant Accidents," adjusted for the then-current licensed power level.

c. The licensee identified that it had derived the source terms used in the EAL threshold calculations from data tabulated in NUREG-1228, rather than the proposed AST. The licensee stated that NUREG-1228 provided that licensing basis evaluations, such as the AST, should not be used to estimate actual accident source terms or offsite doses.

This guidance remains the NRC staff's position, and is therefore applicable. Since the licensee used the source term guidance tabulated in NUREG-1228, rather than the AST, the staff finds the licensee approach to be acceptable for establishing numeric thresholds for radiation monitor EALs.

In the letter dated June 1, 2000 (ADAMS Accession No. ML003724441), the NRC staff approved a 5 percent power uprate. The NUREG-1228 source term was re-adjusted to reflect the uprated licensed power level. This source term was used to calculate the radiation effluent monitor that would correspond to offsite doses of 1 rem TEDE and/or 5 rem Child Thyroid committed dose equivalent. The adjusted NUREG-1228 source terms were also used in establishing the EAL threshold for the containment radiation monitor.

Since the Perry EALs are based on NUREG-1228 guidance, rather than the licensing basis source term, the EAL thresholds are not impacted by the proposed change in the licensing basis source term.

The licensee's documentation in Section 1.0 of Addendum 4 of the submittal identified changes in various plant parameters used as analysis inputs. The licensee's calculation approach for the effluent radiation monitors and containment radiation monitors EAL thresholds appropriately eliminates these parameters from consideration in the threshold calculations. Accordingly, these analysis changes have no impact on the numeric EAL thresholds

The NRC staff has reviewed the licensee's technical basis for the proposed changes. Based on the above evaluation, the NRC staff has determined that the licensee's emergency plan, as modified, will continue to provide reasonable assurance that the licensee can and will take adequate protective measures in the event of a radiological emergency. Therefore, the NRC staff concludes that the licensee's emergency plan, modified as proposed, will continue to meet planning standards of 10 CFR 50.47(b) and the requirements of Appendix E to 10 CFR Part 50.

3.6 Environmental Qualification

Perry previously received NRC approval for implementation of AST methodology for radiological dose calculations for the design basis LOCA in a pilot plant application prior to the issuance of 10 CFR 50.67 and for the FHA as described in RG 1.183 pursuant to 10 CFR, Section 50.67. In this request, the licensee performed new calculations to support full implementation of an AST for each of the applicable design basis events addressed by RG 1.183 using a source term GNF-2 fuel except for the MSLBOC event. The MSLBOC event does not involve postulated fuel failures. The licensee will replace the currently used GE-14 fuel bundles with the GNF-2 fuel beginning with Cycle 16.

The NRC staff focused on impacts to the safety-related electrical systems and the environmental qualification (EQ) of electrical equipment due to the full implementation of the AST. Based on a review of the LAR, the staff requested the licensee to address whether any nonsafety-related systems and components are credited in the AST analyses and describe their impact on the safety-related electrical systems. The licensee stated in its letter dated July 22, 2014, that no nonsafety-related systems and components are credited to reduce doses in the AST analyses. The licensee further clarified that the normal control room ventilation system (CRVS) is conservatively assumed in several event dose analyses to continue to run post-accident for various lengths of time, rather than being isolated/shutdown during the event, to increase the amount of unfiltered inleakage into the control room. The normal CRVS is, therefore, not credited to reduce doses in the AST analyses.

The NRC staff further requested the licensee to address whether any loads are being added to the Perry emergency diesel generators (EDGs). The licensee stated in its letter dated July 22, 2014, that no new loads were added to the EDGs as a result of full implementation of the AST. Based on the above, the NRC staff determined that there is no impact on safety-related electrical systems as a result of full implementation of the AST.

The NRC staff also reviewed the EQ assessment provided in Section 3 of the submittal. The licensee stated that Perry will maintain the TID-based assumptions used to determine the radiation doses in the existing equipment EQ analyses. As stated in Position 6 of RG 1.183, the licensee may use either the AST or the TID-14844 assumptions for performing the required EQ analyses until such time as a generic issue related to the effect of increased cesium releases on EQ doses is resolved. This generic issue has been resolved in a memo dated April 30, 2001 (ADAMS Accession No. ML011210348) and in Supplement 25 to NUREG-0933, Generic Issue 187. The NRC staff concluded in the memo and NUREG-0933 that there was no clear basis for back-fitting the requirement to modify the design basis for EQ to adopt the AST and there would be no discernable risk reduction associated with such a requirement. Therefore, in view of the cited references, the NRC staff finds that it is acceptable for the TID-based assumptions to remain the licensing basis for equipment EQ analyses for Perry.

The NRC staff requested the licensee to discuss potential changes to the EQ profiles as a result of full implementation of the AST, which is based on a source term for GNF-2 fuel. In its letter dated July 22, 2014, the licensee provided a summary conclusion of a calculation that was performed to determine the impacts of the post-accident radiation doses for GNF-2 fuel on equipment EQ. The licensee determined from the calculation that the post-accident doses (maximum and integrated doses) from the GNF-2 source term trended closely with that from the original source term, with the GNF-2 fuel post-accident doses being slightly higher. The increases represent the expected change in the post-accident dose profiles as a result of the transition to the GNF-2 fuel. The licensee determined from a review of the existing equipment EQ analyses that the expected increased doses resulting from transition to the GNF-2 fuel are bounded by the existing EQ doses. Based on this information, the NRC staff finds that the current EQ of electrical equipment will remain bounding for the purpose of implementation of the proposed AST.

Based on the above evaluation, the NRC staff finds the proposed amendment to the Perry licensing bases and TS changes will not impact the licensee's compliance with 10 CFR, Sections 50.49 and 50.67, and GDC 17 requirements and are consistent with the guidance in RG 1.183. Therefore, the NRC staff finds the proposed changes acceptable.

3.7 Review Conclusion

As described above, the NRC staff reviewed the assumptions, inputs, and methods used by the licensee to assess the radiological consequences of DBAs with full implementation of an AST at Perry. The staff finds that Perry used analysis methods and assumptions consistent with the conservative guidance of RG 1.183. The NRC staff finds that the proposed changes 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 proposed LAR is acceptable with respect to the radiological consequences of the analyzed DBAs and the atmospheric dispersion analyses that support those evaluations.

The NRC staff compared the doses estimated by Perry to the applicable acceptance criteria and to the results estimated by the NRC staff in its confirmatory calculations. The NRC staff finds that the licensee's estimates of the TEDE due to DBAs will comply with the requirements of 10 CFR 50.67 and the guidance of RG-1.183. The NRC staff finds the methods and assumptions as wells the estimates of the EAB, LPZ, and CR doses used by Perry to be in compliance with applicable requirements. Therefore, the NRC staff finds that Perry's use of a revised accident source term in its design basis radiological analyses 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 full implementation of AST using GNF-2 fuel is acceptable.

This licensing action is considered a full implementation of the AST. With this approval, the previous accident source term in the Perry design basis is superseded by the AST proposed by the licensee and approved by the associated amendment. The previous offsite and CR accident dose criteria should continue to reflect the TEDE criteria of 10 CFR Part 50.67, or fractions thereof, as defined in RG 1.183.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Ohio State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

This 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 effluent 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 this amendment involves no significant hazards consideration and there has been no public comment on such finding (79 FR 21298; April 15, 2014). Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR, Section 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

6.0 CONCLUSION

The NRC staff 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 this amendment will not be inimical to the common defense and security or to the health and safety of the public.

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

Design Basis Accidents	Location EAB [2]	LPZ [3]	CR	TSC
Loss of Coolant Accident	21.2	6.9	3.0	2.1
Acceptance Criteria	25	25	5	5
Main steamline break accident [4]	5.90×10^{-2}	6.59×10^{-3}	3.11×10^{-2}	[8]
Acceptance Criteria	2.5	2.5	5	
Main steamline break accident [5]	1.17	1.31×10^{-1}	6.22×10^{-1}	[8]
Acceptance Criteria	25	25	5	
Control Rod Drop Accident [6]	1.61×10^{-1}	1.62×10^{-1}	2.63×10^{-1}	[8]
Acceptance Criteria	6.3	6.3	5	
Control Rod Drop Accident [7]	2.80×10^{-6}	2.96×10^{-6}	1.31×10^{-6}	[8]
Acceptance Criteria	6.3	6.3	5	

[1] Total effective dose equivalent
[2] Exclusion area boundary
[3] Low population zone
[4] Maximum RCS equilibrium iodine activity - 0.2 microcuries/gm
[5] Pre-accident iodine spike - 4.0 microcuries/gm
[6] With LOOP - 1376 failed rods at full power
[7] No LOOP - Offgas System operable - 1376 failed rods at full power
[8] TSC dose only calculated for the LOCA which is considered to be the bounding analysis

Table 2-1
Perry CR Atmospheric Dispersion Factors

Time Interval	χ/Q Value (sec/m^3)
0-2 hours	3.5×10^{-4}
2-8 hours	3.5×10^{-4}
8-24 hours	2.1×10^{-4}
1-4 days	1.1×10^{-4}
4-30 days	5.75×10^{-5}

Table 2-2
Perry TSC Atmospheric Dispersion Factors

Time Interval	χ/Q Value (sec/m^3)
0-2 hours	2.6×10^{-4}
2-8 hours	2.6×10^{-4}
8-24 hours	1.6×10^{-4}
1-4 days	8.1×10^{-5}
4-30 days	4.25×10^{-5}

Table 2-3
Perry Off-Site Atmospheric Dispersion Factors

Time Interval	EAB χ/Q Value (sec/m^3)	LPZ χ/Q Value (sec/m^3)
0-2 hours	4.3×10^{-4}	4.8×10^{-5}
2-8 hours		4.8×10^{-5}
8-24 hours		3.3×10^{-5}
1-4 days		1.4×10^{-5}
4-30 days		4.1×10^{-6}

Table 3
Perry AST CR and TSC Data and Assumptions

CR habitability envelope total volume	390,000 ft^3
CR unfiltered inleakage assumptions for LOCA analysis	
t=0 to t=30 minutes	6,600 cfm
t=30 min to 30 days	1,375 cfm
CR unfiltered inleakage assumptions for CRDA & MSLBOC analyses[1]	6,600 cfm
CRERS Actuation	
LOCA	Manual 30 minutes
CRDA[1]	Not credited
MSLBOC[1]	Not credited
TSC[2] habitability envelope total volume	113,412 ft^3
TSC HVAC Flow Rate	37,000 cfm
TSC unfiltered Inleakage t=60 min to 30 days	39.2 cfm
TSC recirculation flow rate	6,000 cfm
TSC Emergency Recirculation System Actuation	Manual 60 minutes
CR and TSC[2] HEPA filter efficiency credited in the dose analysis	
LOCA	99 percent
CRDA	Not credited
MSLBOC	Not credited
CR and TSC[2] Charcoal filter efficiency credited in the dose analysis	
LOCA	80 percent
CRDA	Not credited
MSLBOC	Not credited
CR and TSC breathing rate	
0 - 720 hours	3.5E–04 m^3/sec
CR and TSC occupancy factors	
0 - 24 hours	1
24 - 96 hours	0.6
96 - 720 hours	0.4

[1] The unfiltered inleakage and CRERS assumptions for the CRDA and the MSLBOC correspond to the reported doses in Table 1 which assume the CRERS is not initiated.
[2] TSC dose only calculated for the LOCA, which is considered to be the bounding analysis.

Table 4
Perry AST Data and Assumptions for the LOCA

Rated thermal power (RTP)	3758 MWt
Power level used in LOCA analysis	3833 MWt (102 percent of RTP)
Drywell free volume	2.765×10^5 ft^3
Containment free volume	1.1654×10^6 ft^3
Volume of sprayed region	4.812×10^5 ft^3
Volume of unsprayed region	6.842×10^5 ft^3
Containment spray initiation	Manual at t = 30 minutes
Containment spray average fall height	54.05 ft
Containment and dry well flow rates	
Drywell to unsprayed containment region	
0 - 0.5 hours	0 cfm
0.5 - 2 hours	3000 cfm
2 – 720 hours	2.77×10^5 cfm
Unsprayed containment region to drywell	
0 – 2 hours	0 cfm
2 – 720 hours	2.77×10^5 cfm
Primary containment leak rate	
0 - 24 hours	0.2 percent/day (L_a)
24 - 720 hours	0.69 L_a
Containment leakage into annulus	0.8992 L_a (reduced by 69 percent after 24 hours)
Containment leakage bypassing annulus	0.1008 L_a (reduced by 69 percent after 24 hours)
AEGT system flow to the environment	2000 cfm (recirculation not credited)
AEGT HEPA filter efficiency credited	99 percent removal
AEGT charcoal efficiency credited	0 percent (no credit taken)
Drywell natural deposition	10th percentile Powers Model
Primary containment natural deposition	10th percentile Powers Model
Iodine chemical form in containment atmosphere	
Cesium iodide	95 percent
Elemental iodine	4.85 percent
Organic iodide	0.15 percent
Containment sump pH	≥ 7

Table 4 (continued)
Perry AST Data and Assumptions for the LOCA

Minimum post-LOCA suppression pool volume	122,900 ft^3
Maximum post-LOCA suppression pool temp	< 212 °F
Chemical form of iodine in ESF leakage	
Elemental iodine	97 percent
Organic iodide	3 percent
ESF Leakage flow rate assumptions	
0 - 24 hours	15 gph
24 - 24.5 hours	15 gph plus 50 gpm
24.5 - 720 hours	15 gph
ESF flash fraction	10 percent
MSIV leak rate total	250 scfh - max 100 scfh per line
Broken line leakage (MSL 1)	100 scfh (reduced by 69 percent after 24 hours)
Intact line leakage	150 scfh (reduced by 69 percent after 24 hours)

Main Steam Line Aerosol Removal Fractions

Time after release (hours)	MSL 1 (failed steam line)	MSL 2 (pipe to intact steamlines)	MSL 3 (intact steamline)
0.0	0.681	0.7206	0.714
0.5	0.835	0.7206	0.813
1.5	0.8713	0.7206	0.8361
3.0	0.89	0.7206	0.8449
5.0	0.8614	0.7206	0.8339
7.0	0.8185	0.7206	0.7998
9.0	0.769	0.7206	0.7558
11.0	0.3653	0.7206	0.3807
720	0	0	0

Main Steam Line Removal Efficiencies for Elemental Iodine and Organic Iodide

	MSL 1 (failed steam line)	MSL 2 (pipe to intact steamlines)	MSL 3 (intact steamline)
Elemental Iodine	0.45	0.45	0.45
Organic Iodide	No credit taken	No credit taken	No credit taken

Table 5
PERRY Data and Assumptions for the CRDA

Power Level	Plant startup, at low power after operation at 3,833 MWt
Radial peaking factor	2.0
Number of assemblies in the core	748
Number of equivalent fuel rods per assembly	85.6
Number of fuel bundles experiencing rod clad damage	16
Number of fuel rods experiencing clad damage	1376
Fraction of damaged rods experiencing fuel melt	0.0077
Fraction of fission product in the fuel gap	
Noble gases	0.1
Iodines	0.1
Alkali metals	0.12
Fraction of activity in melted regions released to RCS[1]	
Noble gases	1.0
Iodines	0.5
Alkali metals	0.25
Fraction of activity released to RCS reaching condenser	
Noble gases	1.0
Iodines	0.1
Alkali metals	0.01
Fraction of activity released from condenser[2]	
Noble gases	1.0
Iodines	0.1
Alkali metals	0.01
Release rate from condenser	1 percent per day for 24 hours

[1] No credit is taken for partitioning in the pressure vessel or for removal by the steam separators.
[2] No credit is taken for dilution or holdup within the turbine building. Radioactive decay during holdup in the turbine and condenser is credited.

Table 6
Perry AST Data and Assumptions for the MSLBOC

Noble gas release[1]	TS maximum offgas release rate 358 mCi/sec after 30 min of decay
Maximum equilibrium iodine RCS concentration	0.2 µCi/gm DE I-131(Case 1)
Pre-accident iodine spike RCS concentration	4.0 µCi/gm DE I-131(Case 2)
MSIV isolation time	6.05 seconds
Liquid release	127,376 lbm
Steam release	14,311 lbm
CR isolation and CRERS initiation	Not Credited

Total Curies Released in MSLBOC Dose Calculation

Isotope	0.2 µCi/g DEI (case 1)	4.0 µCi/g DEI (case 2)
I-131	2.630×10^{0}	5.259×10^{1}
I-132	2.639×10^{1}	5.277×10^{2}
I-133	1.819×10^{1}	3.637×10^{2}
I-134	5.739×10^{1}	1.148×10^{3}
I-135	2.632×10^{1}	5.263×10^{2}
Br-83	3.075×10^{0}	6.151×10^{1}
Br-84	4.913×10^{0}	1.337×10^{2}
Kr-83m	1.028×10^{-1}	1.028×10^{-1}
Kr-85m	2.261×10^{-1}	2.261×10^{-1}
Kr-85	8.447×10^{-4}	8.447×10^{-4}
Kr-87	6.740×10^{-1}	6.740×10^{-1}
Kr-88	6.697×10^{-1}	6.697×10^{-1}
Xe-131m	8.582×10^{-4}	8.582×10^{-4}
Xe-133m	9.007×10^{-3}	9.007×10^{-3}
Xe-133	2.581×10^{-1}	2.581×10^{-1}
Xe-135m	9.974×10^{-1}	9.974×10^{-1}
Xe-135	7.042×10^{-1}	7.042×10^{-1}
Xe-138	2.840×10^{0}	2.840×10^{0}

[1]The noble gas release is based on the maximum allowable TS offgas release rate of 358 mCi/sec after 30 minutes of decay. However, the dose analysis takes no credit for decay as the MSLBOC is analyzed as an instantaneous release.

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

Sincerely,

/RA/

Eva A. Brown, Senior Project Manager
Plant Licensing III-2 and
Planning and Analysis Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-440

Enclosures:
1. Amendment No. 166 to NPF-58
2. Safety Evaluation

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