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

October 21, 2009

Vice President, Operations
Arkansas Nuclear One
Entergy Operations, Inc.
1448 S.R. 333
Russellville, AR 72802

SUBJECT: ARKANSAS NUCLEAR ONE, UNIT NO. 1 - ISSUANCE OF AMENDMENT RE: USE OF ALTERNATE SOURCE TERM (TAC NO. MD7178)

Dear Sir or Madam:

The Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 238 to Renewed Facility Operating License No. DPR-51 for Arkansas Nuclear One, Unit No. 1 (ANO-1). The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated October 22, 2007, as supplemented by letters dated April 3, August 14, and September 18, 2008, and August 31, 2009.

The amendment modifies requirements of TS 3.4.12, "RCS [reactor coolant system] Specific Activity," and TS 3.7.4, "Secondary Specific Activity," as related to the use of an alternate source term (AST) associated with accident offsite and control room dose consequences. Implementation of the AST supports adoption of the control room envelope habitability controls in accordance with NRC-approved TS Task Force (TSTF) Standard Technical Specification change traveler TSTF-448, Revision 3, "Control Room Habitability."

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

Sincerely,

Alan Wang for

N. Kaly Kalyanam, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-313
Enclosures:
1. Amendment No. 238 to DPR-51
2. Safety Evaluation

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

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-313

ARKANSAS NUCLEAR ONE, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 238
Renewed License No. DPR-51

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

   A. The application for amendment by Entergy Operations, Inc. (the licensee), dated October 22, 2007, as supplemented by letters dated April 3, August 14, and September 18, 2008, and August 31, 2009, 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 license 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 Renewed Facility Operating License No. DPR-51 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 238, are hereby incorporated in the renewed license. EOI shall operate the facility in accordance with the Technical Specifications.

3. The license amendment is effective as of its date of issuance shall be implemented within 180 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility
Operating License No. DPR-51
and Technical Specifications

Date of Issuance: October 21, 2009

ATTACHMENT TO LICENSE AMENDMENT NO. 238

RENEWED FACILITY OPERATING LICENSE NO. DPR-51

DOCKET NO. 50-313

Replace the following pages of the Renewed Facility Operating License No. DPR-51 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.

Operating License

REMOVE	INSERT
3	3

Technical Specifications

REMOVE	INSERT
3.4.12-1	3.4.12-1
3.4.12-2	3.4.12-2
3.7.4-1	3.7.4-1

(5) EOI, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components;

(6) EOI, pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

c. This renewed license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

EOI is authorized to operate the facility at steady state reactor core power levels not in excess of 2568 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 238, are hereby incorporated in the renewed license. EOI shall operate the facility in accordance with the Technical Specifications.

(3) Safety Analysis Report

The licensee's SAR supplement submitted pursuant to 10 CFR 54.21(d), as revised on March 14, 2001, describes certain future inspection activities to be completed before the period of extended operation. The licensee shall complete these activities no later than May 20, 2014.

(4) Physical Protection

EOI shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans, including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contains Safeguards Information protected under 10 CFR 73.21, is entitled: "Arkansas Nuclear One Physical Security Plan, Training and Qualifications Plan, and Safeguards Contingency Plan," as submitted on May 4, 2006.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.12 RCS Specific Activity

LCO 3.4.12 The specific activity of the reactor coolant shall be:

a. ≤ 1.0 μCi/gm DOSE EQUIVALENT I-131; and

b. ≤ 72/$\bar{E}$ μCi/gm total.

APPLICABILITY: MODES 1 and 2,
MODE 3 with RCS average temperature (T_{avg}) ≥ 500°F.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Specific activity not within limits.	A.1 ----------NOTE---------- LCO 3.0.4.c is applicable. ---------- Restore specific activity to within limit(s).	24 hours
B. Required Action and associated Completion Time not met. OR DOSE EQUIVALENT I-131 > 60 μCi/gm.	B.1 Be in MODE 3 with T_{avg} < 500°F.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.12.1	Verify reactor coolant gross specific activity ≤ 72/$\bar{E}$ μCi/gm.	7 days

SURVEILLANCE		FREQUENCY
SR 3.4.12.2	----------NOTE---------- Only required to be performed in MODE 1. ---------- Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 1.0 μCi/gm.	14 days
SR 3.4.12.3	----------NOTE---------- Not required to be performed until 31 days after a minimum of 2 EFPD and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours. ---------- Determine $\bar{\bar{E}}$.	184 days

3.7 PLANT SYSTEMS

3.7.4 Secondary Specific Activity

LCO 3.7.4 The specific activity of the secondary coolant shall be ≤ 0.1 μCi/gm DOSE EQUIVALENT I-131.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Specific activity not within limit.	A.1 Be in MODE 3.	6 hours
	AND	
	A.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.4.1	Verify the specific activity of the secondary coolant is ≤ 0.1 μCi/gm DOSE EQUIVALENT I-131.	31 days

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 238 TO

RENEWED FACILITY OPERATING LICENSE NO. DPR-51

ENTERGY OPERATIONS, INC.

ARKANSAS NUCLEAR ONE, UNIT NO. 1

DOCKET NO. 50-313

1.0 INTRODUCTION

By application dated October 22, 2007 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML073030537), as supplemented by letters dated April 3, August 14, and September 18, 2008, and August 31, 2009 (ADAMS Accession Nos. ML081000590, ML082280106, ML082620702, and ML092530221, respectively), Entergy Operations, Inc. (the licensee), requested changes to the Technical Specifications (TSs) for Arkansas Nuclear One, Unit No. 1 (ANO-1). The supplemental letters dated April 3, August 14, and September 18, 2008, and August 31, 2009, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on December 18, 2007 (72 FR 71708).

The proposed changes would revise the requirements of TS 3.4.12, "RCS [reactor coolant system] Specific Activity," and TS 3.7.4, "Secondary Specific Activity," as related to the use of an alternative source term (AST) associated with accident offsite and control room dose consequences. In addition, the licensee proposed a license condition that would delay implementation of this amendment pending verification that the steam generators (SGs) are designed to withstand the loading associated with the worst-case large-break loss-of-coolant accident (LBLOCA), as stated in proposed license condition 2.c.(6). Implementation of the AST supports adoption of the control room envelope habitability controls in accordance with NRC-approved TS Task Force (TSTF) Standard Technical Specification change traveler TSTF-448, Revision 3, "Control Room Habitability."

2.0 REGULATORY EVALUATION

Section 182a of the Atomic Energy Act requires applicants for nuclear power plant operating licenses to include TSs as part of the license. The TSs ensure the operational capability of

structures, systems, and components that are required to protect the health and safety of the public. The NRC's regulatory requirements related to the content of the TSs are contained in Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.36, "Technical specifications," which requires that the TSs include items in the following specific categories: (1) safety limits, limiting safety systems settings, and limiting control settings; (2) limiting conditions for operation (LCOs); (3) surveillance requirements (SRs); (4) design features; and (5) administrative controls. In accordance with 10 CFR 50.36(c)(5), administrative controls are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner.

The NRC staff reviewed the proposed changes for compliance with 10 CFR 50.36 and agreement with the precedent as established in NUREG-1430, "Standard Technical Specifications – Babcock and Wilcox [B&W] Plants." In general, licensees cannot justify TS changes solely on the basis of adopting the Standard Technical Specification (STS) model. Licensees may revise the TSs to adopt the improved STS format and content, provided that a plant-specific review supports a finding of continued adequate safety because: (1) the change is editorial, administrative, or provides clarification (i.e., no requirements are materially altered); (2) the change is more restrictive than the licensee's current requirement; or (3) the change is less restrictive than the licensee's current requirement, but nonetheless still affords adequate assurance of safety when judged against current regulatory standards.

The NRC staff reviewed the licensee's evaluation of the radiological consequences of affected design-basis accidents (DBAs) for implementation of the AST methodology, and the associated changes to the TSs proposed by the licensee, against the requirements specified in paragraph (b)(2) of 10 CFR Section 50.67, "Accident source term." The regulations in 10 CFR 50.67(b)(2) require that the licensee's analysis demonstrates with reasonable assurance that:

i. An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 0.25 Sv [Seiverts] (25 rem [roentgen equivalent man]) total effective dose equivalent (TEDE).

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

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

This safety evaluation (SE) addresses the impact of the proposed changes on previously analyzed design-basis accident (DBA) radiological consequences and the acceptability of the revised analysis results. The regulatory requirements from which the NRC staff based its acceptance are the reference values in 10 CFR 50.67, "Accident source term," and the

accident-specific guideline values in Regulatory Position 4.4 of Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Revision 0, July 2000 (ADAMS Accession No. ML003716792), and Table 1, "Accident Dose Criteria," of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (SRP), Section 15.0.1, Radiological Consequence Analyses Using Alternative Source Terms" (ADAMS Accession No. ML003734190). The licensee has not proposed any significant deviation or departure from the guidance provided in RG 1.183. The NRC staff's evaluation is based upon the following regulations, regulatory guides, and standards:

- 10 CFR 50.49, "Environmental qualification of electric equipment important to safety for nuclear power plants," requires that the safety-related electrical equipment which are relied upon to remain functional during and following 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.

- 10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants," requires that preventative maintenance activities must not reduce the overall availability of the systems, structures, or components.

- 10 CFR Section 50.67, "Accident source term," as described above.

- 10 CFR Appendix A of Part 50, General Design Criterion (GDC) 17, "Electric power systems," requires, in part, that nuclear power plants have onsite and offsite electric power systems to permit the functioning of structures, systems, and components that are important to safety. The onsite system is required to have sufficient independence, redundancy, and testability to perform its safety function, assuming a single failure. The offsite power system is required to be supplied by two physically independent circuits that are designed and located so as to minimize, to the extent practical, the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. In addition, this criterion requires provisions to minimize the probability of losing electric power from the remaining electric power supplies as a result of loss of power from the unit, the offsite transmission network, or the onsite power supplies.

- 10 CFR Appendix A of Part 50, GDC 18, "Inspection and testing of electric power systems," requires that electric power systems that are important to safety must be designed to permit appropriate periodic inspection and testing.

- 10 CFR Part 50, Appendix A, GDC 19, "Control room," requires that a control room be provided from which actions can be taken to operate the nuclear reactor safely under normal conditions and to maintain the reactor in a safe condition under accident conditions, including a LOCA. With regard to control room access and occupancy, adequate radiation protection shall be provided to ensure that radiation exposures shall not exceed 0.05 Sv (5 rem) total effective dose equivalent (TEDE), as defined in 10 CFR 50.2 for the duration of the accident.

- 10 CFR 100.11, "Determination of exclusion area, low population zone, and population center distance."

- 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," Revision 3, June 2001 (ADAMS Accession No. ML011710176).

- RG 1.75, "Criteria for Independence of Electrical Safety Systems," Revision 3, dated February 2005 (ADAMS Accession No. ML043630448), describes a method acceptable to the NRC staff for complying with the NRC's regulations with respect to the physical independence requirements of the circuits and electric equipment that comprise or are associated with safety systems.

- RG 1.183 provides guidance to licensees of operating power reactors on acceptable applications of ASTs; the scope, nature, and documentation of associated analyses and evaluations; consideration of impacts on analyzed risk; and content of submittals. This guide establishes an acceptable AST and identifies the significant attributes of other ASTs that may be found acceptable by the NRC staff. This guide also identifies acceptable radiological analysis assumptions for use in conjunction with the accepted AST. This RG states that licensees may use either the AST or the Technical Information Document (TID)-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," U.S. Atomic Energy Commission, March 23, 1962 (ADAMS Legacy Library Accession No. 8202010067), assumptions for performing the required environmental qualification analyses to show that the equipment remains bounding. RG 1.183 further states that no plant modifications are required to address the impact of the difference in source term characteristics (i.e., AST versus TID-14844) on environmental qualification doses. RG 1.183 also states that maintaining pH basic will minimize re-evolution of iodine from the suppression pool water.

- RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," Revision 0, June 2003 (ADAMS Accession No. ML031530505).

- RG 1.196, "Control Room Habitability at Light-Water Nuclear Power Reactors," Revision 0, May 2003 (ADAMS Accession No. ML031490611).

- NUREG-0409, "Iodine Behavior in a PWR Cooling System Following a Postulated Steam Generator Tube Rupture Accident," May 1985.

- SRP Section 2.3.4, "Short-Term Atmospheric Dispersion Estimates for Accident Releases," Revision 3, March 2007 (ADAMS Accession No. ML070730398).

- SRP Section 6.4, "Control Room Habitability Systems," Revision 3, March 2007 (ADAMS Accession No. ML070550069).

- SRP Section 6.5.2, "Containment Spray as a Fission Product Cleanup System," Revision 4, March 2007 (ADAMS Accession No. ML070190178).

- SRP Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," Revision 0, July 2000 (ADAMS Accession No. ML003734190).

- SRP Section 15.6.2, "Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment," Revision 2, July 1981 (ADAMS Accession No. ML052350147).

- NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," February 1995 (ADAMS Accession No. ML0410400063).

- NUREG/CR-5950, "Iodine Evolution and pH Control," December 1992 (ADAMS Accession No. ML063460464).

The NRC staff also considered relevant information in the ANO-1 Safety Analysis Report (SAR) and TSs.

3.0 TECHNICAL EVALUATION

Implementation of the AST by the licensee required re-analyzing several DBAs using new source terms. The licensee performed these tasks by following the requirements of 10 CFR Section 50.67, "Accident source term." The licensee also applied for a license amendment under 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit." An acceptable accident source term is a permissible amount of radioactive material that could be released to the containment from the damaged core following an accident. As a result of improved understanding of the mechanisms of the release of radioactivity, the current accident source term prescribed in 10 CFR 50.67 could be replaced by a less restrictive AST. However, this replacement is subject to performing a successful re-evaluation of the major DBAs. The guidance for implementation of an AST is provided in RG 1.183.

The DBA dose consequence analyses evaluated the integrated TEDE dose at the exclusion area boundary (EAB) for the worst 2-hour period following the onset of the accident. The integrated TEDE doses at the outer boundary of the low-population zone (LPZ) and the integrated dose to an ANO-1 control room operator were evaluated for the duration of the accident. The dose consequence analyses were performed by the licensee using the NRC-sponsored radiological consequence computer code, "RADTRAD: Simplified Model for RADionuclide Transport and Removal And Dose Estimation," Version 3.03, as described in NUREG/CR-6604, "NUREG/CR-6604, 'RADTRAD: A Simplified Model for RADionuclide Transport and Removal And Dose Estimation'," April 1998, including Supplement 1 (dated June 1999) and Supplement 2 (October 2002). The RADTRAD code, developed by the Sandia National Laboratories for the NRC, estimates transport and removal of radionuclides and radiological consequence doses at selected receptors. The NRC staff uses the RADTRAD computer code to perform independent confirmatory dose evaluations as needed to ensure a thorough understanding of the licensee's methods. Although the NRC staff performed its independent radiological consequence dose calculation as a means of confirming the licensee's results, the NRC staff's acceptance is based on the licensee's analyses.

3.1 Atmospheric Dispersion Estimates

3.1.1 Meteorological Data

By letter dated October 1, 2001 (ADAMS Accession No. ML012760524), the licensee stated it used hourly meteorological data previously provided in support of the Arkansas Nuclear One, Unit 2 (ANO-2) power uprate license amendment request (LAR). For the reasons noted in the SE associated with the ANO-2 power uprate Amendment No. 244 dated April 24, 2002 (ADAMS Accession No. ML021130826), the meteorological data, presented for calendar years 1995 through 1999, were found acceptable by the NRC staff evaluation and are considered acceptable for atmospheric dispersion estimates used in the dose assessments performed in support of this LAR.

3.1.2 Control Room Atmospheric Dispersion Factors

The control room atmospheric dispersion factors (χ/Q values) used in the dose assessment for this LAR were generated using the site meteorological data collected from 1995 through 1999 and the ARCON96 atmospheric dispersion computer code (NUREG/CR-6331, Revision 1, "Atmospheric Relative Concentrations in Building Wakes," May 1997). RG 1.194 states that ARCON96 is an acceptable methodology for assessing control room χ/Q values for use in DBA radiological analyses. The NRC staff evaluated the applicability of the ARCON96 model and concluded that there are no unusual siting, building arrangements, release characterization, source-receptor configuration, meteorological regimes, or terrain conditions that would preclude use of this model in support of the amendment for ANO-1.

The licensee stated that the accident scenarios and χ/Q values model the limiting doses considering multiple release scenarios, including those due to loss-of-offsite power or other single failures. Releases were postulated to occur from six release locations and to be transported to both control room intakes. The licensee provided a table of those 12 sets of χ/Q values from which the χ/Q value associated with the limiting source-receptor pair for each time period was identified for each DBA.

- Containment building wall – loss-of-coolant accident (LOCA), control rod ejection accident (CREA), and fuel handling accident (FHA),
- Main steam safety valves (MSSVs) – steam generator tube rupture accident (SGTR),
- Atmospheric dump valves (ADVs) – main steam line break accident (MSLB), SGTR, and CREA,
- Penetration room ventilation system (PRVS) exhaust – LOCA and FHA,
- Main steam line break – MSLB, and
- Releases from the FHA were not limiting in this LAR.

All sources were modeled as ground level releases. The release from the containment was modeled as a diffuse source based upon RG 1.194 guidance and the other releases were modeled as point sources. The licensee conservatively chose worst-case assumptions and release point locations to minimize the distance from the release to each control room intake. The control room intake χ/Q values were used to assess the dose resulting from unfiltered inleakage to the control room. The licensee stated that this was conservative because the calculated χ/Q values were associated with the most direct source-receptor pathways.

In summary, the NRC staff qualitatively reviewed the inputs to the ARCON96 computer runs for the control room χ/Q value assessment and found them consistent with site configuration drawings and NRC staff practice. The NRC staff also reviewed the licensee's dispersion modeling, performed confirmatory calculations and has concluded that the licensee's χ/Q values that are listed in Table 3.1-1 of this SE are acceptable for use in control room DBA dose assessment associated with this amendment request.

3.1.3 Offsite Atmospheric Dispersion Factors

The licensee used the current ANO-1 licensing basis exclusion area boundary (EAB) and low population zone (LPZ) χ/Q values listed in Table 3.1-2 of this SE to assess the radiological consequences of the DBAs postulated in this amendment request. These χ/Q values are presented in Table 14-31 of Amendment No. 22 to the ANO-1 SAR. These χ/Q values were also approved previously by the NRC staff for ANO-1 in an SE dated June 6, 1973, and are still acceptable as the site characteristics have not changed. The NRC staff has concluded that use of these χ/Q values are acceptable for the EAB and LPZ dose estimates.

3.2 Radiological Consequences of DBAs

The licensee has proposed a licensing basis change for its offsite and control room DBA dose consequence analysis for ANO-1. The proposed change will implement an AST methodology for determining DBA offsite and control room doses. For full implementation of the AST DBA analysis methodology, the dose acceptance criteria specified in 10 CFR 50.67 provides an alternative to the previous whole body and thyroid dose guidelines stated in 10 CFR 100.11 and GDC 19. To incorporate a full implementation of the AST, RG 1.183, Regulatory Position 1.2.1 specifies that the DBA LOCA must be reanalyzed.

As stated in RG 1.183, Regulatory Position 5.2, the DBAs addressed in the appendices of RG 1.183 were selected from accidents that may involve damage to irradiated fuel. RG 1.183 does not address DBAs with radiological consequences based on TS reactor or secondary coolant specific activities only. 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.

To support the proposed implementation of an AST, the licensee analyzed the radiological dose consequences of the following DBAs:

- LOCA (SE Section 3.2.1)
- FHA (SE Section 3.2.2)

- MSLB (SE Section 3.2.3)
- SGTR (SE Section 3.2.4)
- CREA (SE Section 3.2.5)

The ANO-1 current licensing basis (CLB) also includes analyses of "loss of load" (LOL), "loss of AC power" (LOAC), "maximum hypothetical accident" (MHA), and "waste gas decay tank rupture" (WGDTR). The loss-of-power cases were not re-evaluated because they are enveloped by the results of the MSLB analysis, which includes consideration of loss-of-offsite power. The CLB MHA and LOCA events are replaced with a single LOCA event. The WGDTR event was not re-evaluated because its analysis uses a source term that is not impacted by the adoption of an AST. Based on the above, the LOAC power and MHA analyses are deleted. The analysis of a LOL event is maintained because this analysis uses 10 CFR Part 20 values, in lieu of 10 CFR Part 100 values, for acceptance criteria. The ANO-1 Calculated Radiological Consequences can be found in Table 3.2-1 of this SE.

The licensee is eliminating its definition for MHA, which describes the worst-case DBA (typically a LOCA with substantial core melt) for dose consequences. Concerned that the licensee's proposed changes may not satisfy NRC regulatory requirements, the NRC staff issued a request for additional information (RAI) in its letter dated August 13, 2008 (ADAMS Accession No. ML082100587), regarding the removal of the MHA and the new design and licensing basis being proposed in the AST analysis. The NRC staff was also concerned with a potential non-compliance issue at B&W nuclear power plants involving the integrity of the SG tubes following an LBLOCA. A loss of tube integrity during an LBLOCA could challenge a B&W nuclear power plant's ability to comply with the NRC's emergency core cooling and radiological consequence regulations. By letter dated August 14, 2008, the licensee responded to the RAI. In the response, the licensee stated that the analyses, on which the Pressurized-Water Reactor Owners Group (PWROG) Topical Report (TR) BAW-2374, Revision 2, "Risk Informed Assessment of Once-Through Steam Generator Loads due to Breaks in Reactor Coolant System Upper Hot Leg Large-Bore Piping," is based, demonstrate that no LOCA will result in any SG tube failures. Therefore, the existence of an AST LOCA source term for a BAW-2374 event that results in SG tube failure is not credible and the bounding AST SGTR source term should be considered applicable to the scenario. The NRC staff concluded that this was not an adequate justification due to the fact that TR BAW-2374, Revision 2, had not been approved by the NRC.

By letter dated August 31, 2009, the licensee proposed to replace the separate MHA and realistic LOCA dose analyses currently in the ANO-1 SAR with a single, bounding AST LOCA analysis. The licensee stated that the worst credible ANO-1 AST accidents for dose consequence purposes are the LOCA for absolute EAB and LPZ doses and the CREA (secondary release case) for control room dose. With the implementation of the AST, the licensee proposed to delete use of the antiquated "Maximum Hypothetical Accident" terminology in the interest of maintaining a clear design basis per the guidance of RG 1.183, Regulatory Position 1.6.

ANO-1's AST analyses assume that steam generator tube integrity is maintained during/ following a LOCA. Since the AST analyses assume tube integrity is maintained during/following a LOCA, the licensee proposed the following license condition:

> In accordance with Technical Specification Amendment 238, the 60-day implementation period associated with the adoption of Alternate Source Term shall be delayed pending verification that the Steam Generators are designed to withstand the loading associated with the worst-case large break Loss of Coolant Accident and that tube integrity will be maintained for this LBLOCA as verified through implementation of Technical Specification 5.5.9, Steam Generator (SG) Program. This verification shall be completed no later than April 1, 2010, and the results of the verification shall be provided to the NRC within 30 days thereafter.

The NRC staff has determined that such a license condition is not necessary since the licensee's steam generator program contained in TS 5.5.9, "Steam Generator (SG) Program," and the associated LCO 3.4.16 require the licensee to ensure that tube integrity is being maintained during all DBAs in order to operate the steam generators. As a result of these TS requirements and the inherent assumptions about tube integrity in the AST LOCA analysis, the NRC staff concludes that the steam generators can only be operated, and the AST can only be implemented, if tube integrity is maintained during/following a LOCA consistent with the assumptions in the AST analyses.

In summary, the TS requirements provide reasonable assurance that the steam generators will be operated consistent with the design and licensing basis established with this amendment request. The NRC will continue to oversee the licensee's implementation of their steam generator program through review of steam generator tube inspection reports submitted in accordance with TS 5.6.7, "Steam Generator Tube Inspection Reports," and through the reactor oversight process.

3.2.1 LOCA Analyses

The LOCA event is assumed to be caused by an abrupt failure of a main reactor coolant pipe and when the emergency core cooling system (ECCS) fails to prevent the core from experiencing significant degradation. This sequence cannot occur unless there are multiple failures and thus goes beyond the typical DBA that considers a single active failure. Activity is released into the containment and then to the environment by means of containment leakage and leakage from the ECCS. This event is described in the Section 14.2.2.5 of the ANO-1 SAR.

For the LOCA analysis, the licensee assumed that the core isotopic inventory that is available for release into the containment atmosphere is based on maximum full power operation of the core at 2619 megawatt thermal (MWt), or 1.02 times the current licensed thermal power level of 2568 MWt. The two percent is added in order to account for the ECCS evaluation uncertainty. Additionally, current licensed values for fuel enrichment and burnup are assumed when determining the core isotopic inventory.

The core inventory release fractions and release timing for the gap and early in-vessel release phases of the DBA LOCA were taken from RG 1.183, Tables 2 and 4, respectively. Also, consistent with RG 1.183 guidance, the licensee assumes that the radioactive iodine speciation

released from failed fuel is 95 percent aerosol (particulate), 4.85 percent elemental, and 0.15 percent organic, whereas the radioactive iodine speciation released from the SGs is 97 percent elemental and 3 percent organic.

The NRC staff has reviewed the licensee's assessment of the following potential post-LOCA activity release pathways:

- Containment leakage directly to atmosphere
- Containment leakage via the penetration rooms
- Emergency safeguards features system (ESFS) leakage into the auxiliary building

3.2.1.1 Containment Leakage Directly to Atmosphere

The ANO-1 containment is projected to leak at its design leakage of 0.2 percent of its contents by weight per day for the first 24 hours and then at 0.1 percent for the remainder of the 30-day accident duration. The licensee uses a two-region containment transport model in assessing the containment leakage pathway. This model is comprised of a region that is sprayed by the containment spray system and an unsprayed region. The sprayed region envelopes 89 percent of the total free volume of the containment. The air in the two regions mixes at a rate equal to two turnovers of the unsprayed region volume per hour and leaks to the environment at the design leakage rate. The licensee did not credit natural deposition as a removal process. The licensee assumes that the sprays become effective at 300 seconds with an aerosol removal coefficient of 2.6 hr^{-1}. The elemental spray coefficient is initially assumed to be 20 hr^{-1} (limit per SRP 6.5.2), but is conservatively further reduced to 10 hr^{-1} at the start of sump recirculation. The licensee limits the elemental iodine decontamination factor (DF) to a maximum value of 200, which was determined to occur at 3.1 hours. No spray removal of elemental iodine is credited after that. Similarly, the licensee decreases that aerosol removal coefficient by a factor of 10 when the aerosol DF reaches 50, which was determined to occur at 3.3 hours. The release of fission products from the containment to the environment occurs as an unfiltered ground level release. The containment leakage pathway modeling is consistent with the guidance in RG 1.183 and SRP 6.5.2.

3.2.1.2 Containment Leakage via the Penetration Rooms

ANO-1 has penetration rooms adjacent to the containment into which 50 percent of the containment leakage is assumed to leak. The remaining 50 percent of the primary containment leakage is assumed to be released directly to the environment as a ground level release without credit for any filtration. These rooms are serviced by a safety-related penetration room ventilation system (PRVS). Leakage from containment collected by the PRVS is processed by engineered safeguards filters prior to release. The PRVS filters and releases postulated post-LOCA leakage through two parallel fans. The fans exhaust through a pipe that runs up the outside of the containment. The pipe is a hooked vent with no vertical velocity.

The PRVS is designed to be in full operation in less than 55 seconds following receipt of a reactor building (containment) isolation signal. Since the onset of the gap release is not

assumed until 30 seconds following a LOCA and the release must then take a tortuous path from the fuel gap to and through the containment wall, the PRVS is assumed to be in operation well before any containment leakage reaches the penetration rooms. Even if some containment leakage does reach a penetration room prior to achieving full PRVS fan flow, it would be held up in the room until sufficient PRVS flow is established. Therefore, it can be assumed that no containment leakage into the penetration rooms is released directly to the environment without filtration.

The PRVS is credited as being capable of maintaining the penetration rooms at a negative pressure with respect to the outside environment throughout the event as described in SAR Section 6.5. No credit is taken for dilution in the penetration room volume. The PRVS is credited as meeting the requirements of RG 1.52. The filters in the PRVS ventilation system are credited at 99 percent efficiency for particulates and 90 percent for both elemental and organic iodine. Based on the above discussion, the assumption that 50 percent of the containment leakage would be into the penetration rooms is conservative and, therefore, acceptable to the NRC staff.

3.2.1.3 Emergency Safeguards Features System (ESFS) Leakage into the Auxiliary Building

ESFSs that re-circulate water outside the primary containment are assumed to leak during their intended operation. With the exception of noble gases, all fission products released from the fuel to the containment are assumed to instantaneously and homogeneously mix in the containment sump water at the time of release from the core.

The reactor building spray and low-pressure injection pumps are located in sealed rooms of the auxiliary building through which air does not circulate. Therefore, iodine leaking from these pumps is not exhausted to the environment. These are the only pumps that recirculate sump water following any LOCA of sufficient size to result in fuel damage. A flow path does exist from these pumps through the penetration rooms and back into containment. Leakage from this flow path outside the sealed rooms is evaluated for its dose impact. No credit for filtration of this leakage by the PRVS is taken. This leakage is assumed to be 782 cubic centimeters per hour (cc/hr), which is two times the leakage limit of 391 cc/hr identified in the ANO-1 SAR. The leakage is assumed to start at the time recirculation flow occurs in these systems and continue for the 30-day duration. The ECCS pumps do not have mini-flow returns to the borated (refueling) water storage tank (BWST) and there is no viable means of leakage of sump fluid to the BWST. There is a single, common return line from the ECCS pump discharge lines to the BWST. In order for any sump fluid to get into the BWST would require the failure of redundant, closed and manually-operated valves in this line.

With the exception of iodine, all radioactive materials in the recirculating fluid are assumed to be retained in the liquid phase. A flashing fraction of 4.58 percent was calculated based upon the sump temperature at the time of recirculation. However, the flashing fraction used in the analysis is limited based on Regulatory Position 5.5 of RG 1.183. Since the calculated flashing fraction is less than 10 percent, the amount of iodine that becomes airborne is conservatively assumed to be 10 percent of the total iodine activity in the leaked fluid. For ECCS leakage into the auxiliary building, the form of the released iodine is 97 percent elemental and 3 percent organic. No reduction in release activity by dilution or holdup within buildings, or by any ventilation system, is credited.

3.2.1.4 Control Room Doses

The radioactivity from the above sources are assumed to be released into the atmosphere and transported to the control room air intake, where it may leak into the control room envelope or be filtered by the control room intake and recirculation filtration system and distributed in the control room envelope. The three major radioactive sources which contribute to the control room TEDE dose are:

- Post-LOCA airborne activity inside the control room
- Post-LOCA airborne cloud external shine to control room
- Post-LOCA containment shine to control room

For this event, the control room ventilation system cycles through two modes of operation: normal and emergency. Initially, the ventilation system is assumed to be operating in its normal mode supplying 35,200 cubic feet per minute (cfm) of unfiltered makeup flow rate. After the start of the event, the control room is assumed to be isolated due to a high radiation signal. This signal may be initiated due to containment-shine, shine from the approaching radioactive cloud, or actual initial entry of radioactive material into the normal ventilation ductwork. A loss-of-offsite power would also initiate control room isolation. Control room isolation is designed to occur within 5 seconds, but a 10-second delay is assumed in the analysis. During isolation of normal control room ventilation, the control room emergency ventilation system (CREVS) automatically starts up and pressurizes the control room. After isolation of normal control room ventilation, 333 cfm of filtered in-leakage is assumed to be supplied by the CREVS. The CREVS is also assumed to recirculate and filter 1,667 cfm of control room air. The CREVS filter efficiencies that are applied to the filtered makeup air are 99 percent for particulate, elemental iodine, and organic iodine and to the recirculation flow are 99 percent for particulate and 95 percent for elemental iodine and organic iodine. The control room unfiltered in-leakage is conservatively assumed to be 82 cfm during the CREVS transition period of 30 minutes after a LOCA.

Gamma ray dose rates inside the control room are calculated using the SCAP-II computer code. The SCAP code is based on the point kernel method for calculation of radiation dose in complex source-shield geometries. In this case, the geometry applied in the SCAP calculation includes simulations of the walls and ceiling of the control room as well as of the outer structure of the reactor containment building. The cylindrical portion of the containment is treated as a 116 feet diameter cylindrical shell with a thickness of 3.75 feet of concrete. The height of the cylinder is considered to be 179 feet. The control room is treated as a structure with concrete wall and ceiling thickness of 1.5 feet. The gamma ray source strengths used in the calculations are determined using the ORIGEN-S computer code. Activity releases following the DBA event are based on the AST scenario defined in NUREG-1465 and RG 1.183. Reactor coolant activity is released during the first 30 seconds after a LOCA followed by a gap release phase during which all of the gap activity (3 percent of the total core inventory of volatile nuclides) is instantaneously released. In addition, for accidents where long-term fuel cooling or core geometry are not maintained, an additional release of 2 percent of the inventory of volatile core inventories are considered to be released at a constant rate over a 30-minute gap release phase. Volatile species are considered to be noble gases, halogens, and alkali metals. Following the gap

release phase, an in-vessel release phase is considered, which lasts for 1.3 hours. These releases are assumed to be at a constant rate over the release interval.

The radioactivity releases and radiations levels used for the control room dose are determined using the same source term, transport, and release assumptions used for determining the EAB and LPZ TEDE values. The NRC staff concludes that the licensee used analyses, assumptions, and methods that are consistent with RG 1.183.

3.2.1.5 LOCA Analyses Conclusion

The licensee evaluated the radiological consequences resulting from the postulated LOCA using the AST and concluded that the radiological consequences at the EAB, LPZ and in the control room are within the dose criteria specified in 10 CFR 50.67. The NRC staff has reviewed the licensee's evaluation. In performing this review, the staff relied upon information provided by the licensee, staff experience in performing similar reviews and, where deemed necessary, confirmatory calculations were performed. The staff reviewed the methods, parameters, and assumptions that the licensee used in its radiological dose consequence analyses and concludes that they are consistent with the conservative guidance provided in RG 1.183. The LOCA analysis assumption and parameters can be found in Table 3.2-2 of this SE. The NRC staff concludes that the revised LOCA analyses using the AST meets the relevant dose acceptance criteria and is, therefore, acceptable with the respect to the radiological consequences of DBAs.

3.2.2 FHA Analyses

This accident analysis postulates that a spent fuel assembly is dropped during refueling, damaging 82 fuel rods (six rows of rods in one assembly). According to the licensee's April 3, 2008, letter, the FHA analysis considers both a dropped fuel assembly inside the containment with the equipment hatch open and an assembly drop in the FHA. In both cases, the release is assumed to occur directly to the environment without filtration. The assumptions chosen for this evaluation are appropriate for either location. The core inventory is assumed to decay for 72 hours, the earliest that fuel movement will commence. The radial peaking factor of 1.8 was conservatively used to represent the most limiting fuel assembly. The entire gap inventory in the damaged rods is assumed to be release from the fuel. The licensee assumed that 12 percent of the I-131 inventory of the core was in the fuel rod gap (modified per NUREG/CR-5009, "Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors," February 1988), along with 10 percent of the Kr-85 and 10 percent of all other iodines and noble gases. These assumptions are more conservative than those presented in RG 1.183 and are, therefore, acceptable. Alkali metals make a negligible contribution to dose for this analysis and were omitted.

Fission products released from the damaged fuel are decontaminated by passage through the overlaying water in the reactor cavity or spent fuel pool depending on their physical and chemical form. Following the guidance in RG 1.183, Appendix B, Regulatory Position 1.3, the licensee assumed that the chemical form of radioiodine released from the fuel to the spent fuel pool consists of 95 percent cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. The CsI released from the fuel is assumed to completely dissociate in the pool water, and because of the low pH of the pool water, the iodine CsI re-evolves as elemental

iodine. This results in a final iodine distribution of 99.85 percent elemental iodine and 0.15 percent organic iodine. The licensee assumed that the release to the pool water and the chemical redistribution of the iodine species occurs instantaneously.

The licensee assumed the depth of water above the damaged fuel is 23 feet. Consistent with RG 1.183, Appendix B, Regulatory Position 2, as corrected by item 8 of NRC Regulatory Issue Summary (RIS) 2006-04, "Experience with Implementation of Alternative Source Terms," March 7, 2006 (ADAMS Accession No. ML053460347), the licensee assumed that the decontamination factors for the elemental and organic species are 285 and 1, respectively, giving an overall effective decontamination factor of 200 (i.e., 99.5 percent of the total iodine released from the damaged rods is retained by the water). This difference in decontamination factors for elemental (99.85 percent) and organic iodine (0.15 percent) species results in the iodine above the water being composed of 70 percent elemental and 30 percent organic species. Consistent with RG 1.183, the licensee credited an infinite DF for the remaining particulate forms of the radionuclides contained in the gap activity. In accordance with RG 1.183, the licensee did not credit decontamination from water scrubbing for the noble gas constituents of the gap activity.

For this event, the control room ventilation system cycles through two modes of operation: normal and emergency. Initially, the ventilation system is assumed to be operating in its normal mode supplying 35,200 cubic feet per minute (cfm) of unfiltered makeup flow rate. After the start of the event, the control room is assumed to be isolated due to a high radiation signal. This signal may be initiated due to containment-shine, shine from the approaching radioactive cloud, or actual initial entry of radioactive material into the normal ventilation ductwork. A loss-of-offsite power would also initiate control room isolation. Control room isolation is designed to occur within 5 seconds, but a 10-second delay is assumed in the analysis. During isolation of normal control room ventilation, the control room emergency ventilation system (CREVS) automatically starts up and pressurizes the control room. After isolation of normal control room ventilation, 333 cfm of filtered in-leakage is assumed to be supplied by the CREVS. The CREVS is also assumed to recirculate and filter 1,667 cfm of control room air. The CREVS filter efficiencies that are applied to the filtered makeup air are 99 percent for particulate, elemental iodine, and organic iodine and to the recirculation flow are 99 percent for particulate and 95 percent for elemental iodine and organic iodine. The control room unfiltered in-leakage is conservatively assumed to be 82 cfm during the CREVS transition period of 30 minutes after a LOCA.

The licensee assumed that the containment would not be isolated and that the FHA ventilation system would not be credited. The normal air intake is assumed to continue for the 30-day accident duration.

3.2.2.1 FHA Analyses Conclusion

The licensee evaluated the radiological consequences resulting from the postulated FHA using the AST and concluded that the radiological consequences at the EAB, LPZ, and in the control room are within the dose criteria specified in 10 CFR 50.67. The NRC staff has reviewed the licensee's evaluation. In performing this review, the NRC staff relied upon information provided by the licensee; NRC staff experience in performing similar reviews; and, where deemed necessary, confirmatory calculations were performed. The NRC staff reviewed the methods,

parameters, and assumptions that the licensee used in its radiological dose consequence analyses and concludes that they are consistent with the conservative guidance provided in RG 1.183. The FHA analysis assumption and parameters can be found in Table 3.3-3 of this SE. The NRC staff concludes that the revised FHA analyses using the AST meets the relevant dose acceptance criteria and is, therefore, acceptable with the respect to the radiological consequences of DBAs.

3.2.3 MSLB

The accident considered is the complete severance of a main steam line outside the containment but upstream of the main steam isolation valves. Upon a MSLB, the affected SG rapidly depressurizes. The rapid secondary depressurization causes a reactor power transient, resulting in a reactor trip. Plant cooldown is achieved via the remaining unaffected SG. The radiological consequences of a break outside containment will bound the results from a break inside containment. The MSLB results in the secondary water in the affected once-through steam generator (OTSG) to completely flash to steam. It is assumed that there is a primary-to-secondary leak that allows reactor coolant to leak into the affected OTSG at a rate of 0.5 gallons per minute (gpm), which is half of the maximum leak rate allowed by TS 5.5.9. No fuel damage is postulated for this event.

In the faulted SG, all of the primary-to-secondary leakage is assumed to flash to vapor and be released to the environment with no mitigation. For the unaffected SG used for plant cooldown, a portion of the leakage is assumed to flash to vapor based on the thermodynamic conditions in the reactor and secondary coolant. To address iodine transport for release from the unaffected SG, the licensee developed a steaming model using the flashing fraction of the primary-to-secondary leakage during cooldown as 0.05. The release of the remaining 95 percent of the activity in the leakage is considered in one of two ways: vaporization or mixing with the SG liquid. A portion (calculated to be about 5 percent) of the primary-to-secondary leakage is assumed to be vaporized due to heat transfer across the SG tubes in the steam-covered region of the OTSG. This fraction is thus added to the flashed fraction to provide a total flashing plus vaporization fraction of approximately 0.1. For conservatism, the MSLB analysis assumes this flashing fraction is 0.2. The flashed and vaporized portion of the leakage is assumed to be directly released from the RCS to the atmosphere with no partitioning in the SG. The remaining portion (80 percent) of the primary-to-secondary leakage that is discharged as liquid to the unaffected SG is assumed to be mixed with the SG secondary side liquid inventory and released to the atmosphere with partitioning via steam releases from the bulk fluid in the SG. The SG tubes remain partially covered throughout the event. All values are held constant throughout the duration of the cooldown. The radioactivity within the bulk water is assumed to become vapor at a rate that is a function of the steaming rate and the partition coefficient. A partition coefficient of 100 is assumed for the iodine. The retention of particulate radionuclides in the unaffected SG is limited by the moisture carryover from the SG, which is 0.1 percent. Thus, the partition coefficient for alkali metals is 0.001. No reduction in the release from the faulted SG is assumed.

Two iodine-spiking cases are considered. The first assumes that an iodine spike occurred just before the SGTR and RCS iodine inventory is 60 μCi/gm DE I-131. This is the pre-existing spike case. The second case assumes the event initiates an iodine spike. The spiking model assumes the primary coolant activity is initially at 1.0 μCi/gm DE I-131. Iodine is assumed to be

released from the fuel into the RCS at a rate of 500 times the iodine equilibrium release rate for a period of 8 hours. This is the accident-initiated spike case.

The initial RCS activity is assumed to be at the proposed, revised TS limit of 1.0 µCi/gm DE I-131 and 72/E-bar gross activity. The initial SG activity is assumed to be at the proposed, revised TS limit of 0.1 µCi/gm DE I-131. The steam mass release rates for the intact SG are 2.5325×10^5 grams per minute (gm/min) from 0 to 2 hours, then 3.611×10^5 gm/min from 2 to 237.8 hours after event initiation. These values are based upon a cooldown rate of 100 degrees Fahrenheit per hour (°F/hr) for the first 2 hours followed by a cooldown rate of 4.634 °F/hr until the decay heat removal system is assumed to be placed in service and the intact SG isolated. Cooldown then continues using the decay heat removal system until the RCS temperature is reduced to 212 °F at 251.8 hours, stopping further releases through the faulted SG. This evaluation assumes that the RCS mass remains constant throughout the MSLB event. For the purposes of determining the iodine concentration of the SG secondary, the mass in the unaffected SG is assumed to remain constant throughout the event. Secondary releases from the intact SG are postulated to occur from the atmospheric dump valves (ADVs) using the most limiting atmospheric dispersion factors. Secondary releases from the faulted SG are postulated to occur from main steam pipe using the most limiting atmospheric dispersion factors.

For this event, the control room ventilation system cycles through two modes of operation: normal and emergency. Initially, the ventilation system is assumed to be operating in its normal mode supplying 35,200 cfm of unfiltered, fresh, outside air. After the start of the event, the control room is assumed to be isolated due to a high-radiation signal. This signal may be initiated due to shine from the approaching radioactive cloud or actual initial entry of radioactive material into the normal ventilation ductwork. A loss-of-offsite power (LOOP) would also initiate control room isolation. Control room isolation is designed to occur within 5 seconds, but a 10-second delay is assumed in the analysis. During isolation of normal control room ventilation, the CREVS automatically starts up and pressurizes the control room. After isolation of normal control room ventilation, 333 cfm of filtered, outside makeup air is assumed to be supplied by the CREVS. The CREVS is also assumed to recirculate and filter 1,667 cfm of control room air. The CREVS filter efficiencies that are applied to the filtered makeup air are 99 percent for particulate, elemental iodine, and organic iodine and to the recirculation flow are 99 percent for particulate and 95 percent for elemental iodine and organic iodine.

3.2.3.1 MSLB Analyses Conclusion

The licensee evaluated the radiological consequences resulting from the postulated MSLB using the AST and concluded that the radiological consequences at the EAB, LPZ, and in the control room are within the dose criteria specified in 10 CFR 50.67. The NRC staff has reviewed the licensee's evaluation. In performing this review, the NRC staff relied upon information provided by the licensee, staff experience in performing similar reviews and, where deemed necessary, confirmatory calculations were performed. The NRC staff reviewed the methods, parameters, and assumptions that the licensee used in its radiological dose consequence analyses and concludes that they are consistent with the conservative guidance provided in RG 1.183. The MSLB analysis assumption and parameters can be found in Table 3.2-4 of this SE. The NRC staff concludes that the revised MSLB analyses using the AST meets the relevant dose acceptance criteria and is, therefore, acceptable with the respect to the radiological consequences of DBAs.

3.2.4 SGTR Analyses

The accident considered is the double-ended rupture of a single OTSG tube. The radiological consequences of this event are caused by the transfer of radioactive reactor coolant to the secondary side of the OTSG and the subsequent release of radioactive materials to the environment. No fuel melt or clad breach is postulated for this event.

Following an SGTR, the plant is assumed to continue to operate at full power for 11 minutes until a low RCS pressure reactor trip occurs. All primary-to-secondary leakage, as well as the ruptured tube flow, will be directed to the condenser where it is partitioned prior to release. A loss-of-offsite power is assumed to occur coincident with the reactor trip. The loss-of-offsite power results in a loss of the condenser causing the MSSVs to open and provide steam relief. At 20 minutes, the operators initiate emergency cooldown of the RCS, and then isolate the affected SG at 34 minutes when the RCS temperature has decreased to a value that corresponds to the saturation pressure which is below the lowest MSSV setpoint. At this point, only the unaffected SG is used to continue cooldown to decay heat removal entry conditions and the release point then becomes the ADV with the worst χ/Q values to the control room ventilation intakes. Using only one SG, it will take 237.8 hours to initiate decay heat removal (DHR) and isolate the unaffected SG, thus terminating the release. Primary-to-secondary SG leakage is assumed to be at the ANO-1 TS maximum 150 gallons per day (gpd) per SG throughout the event. The primary-to-secondary leakage is assumed to continue until after the decay heat removal system has been placed in service and both SGs are isolated.

Two iodine-spiking cases are considered. The first assumes that an iodine spike occurred just before the SGTR and RCS iodine inventory is at the proposed, revised 60 μCi/gm DE I-131. This is the pre-existing spike case. The second case assumes the event initiates an iodine spike. The spiking model assumes the primary coolant activity is initially at the proposed, revised TS 3.4.12 value of 1.0 μCi/gm DE I-131. Iodine is assumed to be released from the fuel into the RCS at a rate of 335 times the iodine equilibrium release rate for a period of 8 hours. This is the accident-initiated spike case.

A portion of the primary-to-secondary leakage and ruptured tube flow is assumed to flash to vapor based on the thermodynamic conditions in the reactor and secondary coolant. To address iodine transport for release from the ANO-1 SGs, a steaming model was developed and based on conservative calculations, the flashing fraction of the primary-to-secondary leakage during cooldown is 0.05. The release of the remaining 95 percent of the activity in the leakage is considered in one of two ways: vaporization or mixing with the SG liquid. A portion (calculated to be about 5 percent) of the primary-to-secondary leakage and ruptured tube flow is assumed to be vaporized due to heat transfer across the SG tubes in the steam-covered region of the OTSG. This fraction is then added to the flashed fraction to provide a total flashing plus vaporization fraction of approximately 0.1. For conservatism, the SGTR analysis assumes this flashing fraction is 0.15. The flashed and vaporized portion of the leakage and ruptured tube flow is assumed to be directly released from the RCS to the atmosphere with no partitioning in the SG. The remaining portion (85 percent) of the primary-to-secondary leakage and ruptured tube flow that is discharged as liquid is assumed to be mixed with the SG secondary side liquid inventory and released to the atmosphere with partitioning via steam releases from the bulk fluid in the SG. The SG tubes remain partially covered throughout the event. All values are held

constant throughout the duration of the cooldown. The radioactivity within the bulk water is assumed to become vapor at a rate that is a function of the steaming rate and the partition coefficient. A partition coefficient of 100 is assumed for the iodine. The retention of particulate radionuclides in the SGs is limited by the moisture carryover from the SGs, which is 0.1 percent. Thus, the partition coefficient for alkali metals is 0.001.

For this event, the control room ventilation system cycles through two modes of operation: normal and emergency. Initially, the ventilation system is assumed to be operating in its normal mode supplying 35,200 cfm of unfiltered, fresh, outside air. The control room is assumed to be isolated due to a loss of offsite power at 11 minutes, although a high-radiation signal, due to shine from the approaching radioactive cloud or actual initial entry of radioactive material into the normal ventilation ductwork, is expected to initiate isolation sooner. With isolation of normal control room ventilation, the CREVS starts up and pressurizes the control room. After isolation of normal control room ventilation, 333 cfm of filtered, outside makeup air is assumed to be supplied by the CREVS. The CREVS is also assumed to recirculate and filter 1,667 cfm of control room air. The CREVS filter efficiencies that are applied to the filtered makeup air are 99 percent for particulate, elemental iodine, and organic iodine, and to the recirculation flow are 99 percent for particulate and 95 percent for elemental iodine and organic iodine.

3.2.4.1 SGTR Analyses Conclusion

The licensee evaluated the radiological consequences resulting from the postulated SGTR using the AST and concluded that the radiological consequences at the EAB, LPZ, and in the control room are within the dose criteria specified in 10 CFR 50.67. The NRC staff has reviewed the licensee's evaluation. In performing this review, the staff relied upon information provided by the licensee; staff experience in performing similar reviews and, where deemed necessary, confirmatory calculations were performed. The staff reviewed the methods, parameters, and assumptions that the licensee used in its radiological dose consequence analyses and finds that they are consistent with the conservative guidance provided in RG 1.183. The SGTR analysis assumption and parameters can be found in Table 3.3-5 of this SE. The NRC staff concludes that the revised SGTR analyses using the AST meets the relevant dose acceptance criteria and is, therefore, acceptable with the respect to the radiological consequences of DBAs.

3.2.5 CREA Analyses

This accident analysis postulates the uncontrolled withdrawal of a single control rod. The CREA results in a reactivity insertion that leads to a core power level increase and subsequent reactor trip. Following the reactor trip, plant cooldown is performed using steam release from the SG ADVs. Two CREA cases were considered. The first case assumes that 100 percent of the activity released from the damaged fuel is instantaneously and homogeneously mixed throughout the containment atmosphere. The second case assumes that 100 percent of the activity released from the damaged fuel is completely dissolved in the primary coolant and is available for release to the secondary system. This event is described in ANO-1 SAR Section 14.2.2.4.

The licensee assumed that 14 percent of the fuel rods suffer sufficient damage to result in the release of their entire gap inventory to the RCS and the containment. The licensee assumed

that 10 percent of the core inventory of noble gases and iodines are in the fuel rod gap. The thermo-hydraulic analyses show that fuel melt does not occur and there is no release from the fuel pellets. For the containment leakage case, all of the fission products released from the fuel are assumed to enter the containment. For the secondary release pathway, all of the fission products released from the fuel gap are assumed to remain in the RCS and be available for leakage to the secondary and the environment. A core radial peaking factor of 1.8 is conservatively assumed.

For the containment leakage case, the iodine released is 95 percent CsI, 4.85 percent elemental, and 0.15 percent organic. The licensee assumes the iodine species in the secondary to be 97 percent elemental and 3 percent organic. Sedimentation of particulates in the containment is credited. Containment spray and PRVS are not credited in the CREA analysis.

The containment is projected to leak at its design leakage of the 0.2 percent of its contents by weight per day for the first 24 hours and then at 0.1 percent for the remainder of the 30-day accident durations. The licensee does not credit reduction of iodine by containment sprays. The releases from the containment are released unfiltered to the environment via a ground level release using the limiting containment release point χ/Q.

For the secondary release case, primary coolant activity is released into the SGs by leakage across the SG tubes. The activity on the secondary side is then released via steaming from the MSSVs or ADVs until the decay heat removal system is assumed to be placed into service and the SGs isolated at 38.25 hours into the event. All noble gases associated with this leakage are assumed to be released directly to the environment.

A primary-to-secondary leakage of 300 gpd is assumed. A portion of the primary-to-secondary leakage is assumed to flash to vapor based on the thermodynamic conditions in the reactor and secondary coolant. The SG tubes remain partially covered throughout the event. To address iodine transport for release from the ANO-1 SGs, a steaming model was developed and based on conservative calculations, the flashing fraction of the primary-to-secondary leakage during cooldown is 0.05. This value is held constant throughout the duration of the cooldown. The flashed portion of the primary-to-secondary leakage is modeled as a direct release from the RCS to the environment with no credit for partitioning or depletion.

3.2.5.1 CREA Conclusion

The licensee evaluated the radiological consequences resulting from the postulated CREA using the AST and concluded that the radiological consequences at the EAB, LPZ, and in the control room are within the dose criteria specified in 10 CFR 50.67. The NRC staff has reviewed the licensee's evaluation. In performing this review, the NRC staff relied upon information provided by the licensee, staff experience in performing similar reviews and, where deemed necessary, performed confirmatory calculations. The NRC staff reviewed the methods, parameters, and assumptions that the licensee used in its radiological dose consequence analyses and concludes that they are consistent with the conservative guidance provided in RG 1.183. The CREA analysis assumption and parameters can be found in Table 3.3-6 of this SE. The NRC staff concludes that the revised CREA analyses using the AST meets the

relevant dose acceptance criteria and is, therefore, acceptable with the respect to the radiological consequences of DBAs.

3.2.6 PH Control

After a LOCA, a variety of different chemical species are released from the damaged core. One of them is radioactive iodine. This iodine, when released to the outside environment, will significantly contribute to radiation doses. It is, therefore, essential to keep it confined within the plant's containment. According to NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plant," iodine is released from the core in three different chemical forms; at least 95 percent is released in ionic form as CsI and the remaining 5 percent as elemental iodine (I_2) and hydriodic acid (HI); the release contains at least 1 percent of each I_2 and HI. CsI and HI are ionized in water and are therefore soluble. However, elemental iodine is scarcely soluble.

To sequester the iodine in water, it is desirable to maintain as much as possible of the released iodine in ionic form. In radiation environments existing in containment, some of the ionic iodine dissolved in water is converted into elemental form. The degree of conversion varies significantly with the pH of water. At a higher pH, conversion to elemental form is lower and at pH >7 it becomes negligibly small. The relationship between the rate of conversion and pH is specified in Figure 3.1 of NUREG/CR-5950, "Iodine Evolution and pH Control."

ANO-1 uses sodium hydroxide to control post-LOCA sump pH. In support of a previous license amendment to the ANO-1 TSs regarding a change to the sodium hydroxide concentrations, License Amendment No. 234, dated January 13, 2009 (ADAMS Accession No. ML083050176), the licensee provided information regarding the assumptions and calculations used to verify that the sump pH would remain greater than 7.0 following a LOCA. As part of a response to an NRC staff RAI, the licensee provided this information in a letter dated October 2, 2008 (ADAMS Accession No. ML082770149). The licensee's analysis considered boron concentrations and volumes for the borated water storage tank, core flood tanks, and RCS. Additional inputs included the sodium hydroxide tank concentrations, sodium hydroxide tank volume, and the impact of strong acids generated by radiation of cable insulation and sump water.

The licensee's sump pH analysis used the computer code MULTEQ-REDOX Version 2.24 from the Electric Power Research Institute. Strong acids generated from cable insulation and sump fluids under accident radiation levels were inputs to the code. The bounding strong acid quantities used were 1.52×10^{-3} moles/liter (54 parts per million (ppm)) for hydrochloric acid and 5.85×10^{-5} moles/liter (3.6 ppm) for nitric acid based on a 30-day integrated dose for a LOCA. The minimum pH evaluation used the maximum borated water source volumes and concentrations with the minimum sodium hydroxide tank volume (4,000 gallons) to determine the concentration needed to ensure an equilibrium sump pH greater than 7.0. The calculation determined that a sodium hydroxide tank concentration of 5.53 percent, at the minimum volume, would be sufficient to maintain pH greater than 7.0. Any concentration of sodium hydroxide greater than 5.53 percent will ensure that the sump pool pH will remain in an alkaline regime under the worst-case boron concentrations, sump fluid volumes, and quantities of strong acid generated. The current ANO-1 TS require a minimum sodium hydroxide tank concentration of 6 percent. The TS limit on minimum sodium hydroxide tank concentration will ensure that there is sufficient sodium hydroxide available to maintain the post-LOCA sump pH above 7.0.

The NRC staff reviewed the licensee's assumptions and analysis and concluded that conservative values were used for the key parameters of the calculation. In addition, the staff verified that the current ANO-1 TS requirements for sodium hydroxide tank concentration and volume will ensure sufficient buffering of the sump pool such that the pH will not drop below 7.0.

The NRC staff reviewed the licensee's assumptions to minimize iodine re-evolution as presented in the re-analysis of the radiological consequences for a LOCA. The methodology relies on using the buffering action of sodium hydroxide. The staff concluded that the assumptions are appropriate and consistent with the methods accepted by the staff for the calculation of post-accident containment sump pH and that the post-accident containment sump pH will be maintained above 7.0 for 30 days following a LOCA.

3.2.7 Electrical and Environmental Qualification

The licensee has proposed using an AST to determine accident offsite and control room doses. The licensee stated that the AST involves quantities, isotopic composition, chemical and physical characteristics, and release timing of radioactive material for use as inputs to accident dose analyses.

Upon reviewing the amendment request, the NRC staff requested additional information regarding whether electrical non-safety related systems were credited in the AST analyses. In its letter dated August 31, 2009, the licensee stated that the AST does not rely on any non-safety components that require electrical input.

The NRC staff further requested additional information on whether any loads were being added to the ANO-1 emergency diesel generators (EDGs) and if so, how the loads being added would affect the capability and capacity of the EDGs as well as provide any changes to the loading sequence. In its letter dated August 31, 2009, the licensee stated that no loads were added to the ANO-1 EDGs as a result of the AST adoption.

The NRC staff requested the licensee to provide a list and descriptions of components added to its 10 CFR 50.49 program due to the AST and additionally, confirm that these components are qualified for the environmental conditions to which they are expected to be exposed. In its letter dated August 31, 2009, the licensee stated that no components were added to the ANO-1 10 CFR 50.49 program as a result of the AST adoption.

Lastly, the NRC staff requested additional information regarding any changes in the chemical composition of the chemical spray solution to determine if the components were qualified for their environment. In its letter dated August 31, 2009, the licensee stated that the chemical composition of the ANO-1 chemical spray solution and operation of the chemical spray system are not changed as a result of AST adoption.

The NRC staff also reviewed the environmental qualification portion of the amendment request. The licensee used the methodology contained in TID-14844 to determine the radiation doses in the existing environmental qualification analyses as stated in Attachment 2, Section 2.7 of the licensee's letter dated April 3, 2008. As noted previously, the use of this methodology is consistent with the guidance contained in RG 1.183. Since the licensee will continue to use the TID-14844 methodology and no new equipment is added to its 10 CFR 50.49 program, the

environmental qualification of equipment will remain bounding during full-scope implementation of an AST and is, therefore, acceptable.

3.2.8 Proposed TS Changes

The change to the ANO-1 licensing basis involves the adoption of an AST for calculating accident doses to control room personnel and offsite receptors. The licensee requested the following TS revisions to support assumptions associated with the new AST analyses:

3.2.8.1 TS 3.4.12, "RCS Specific Activity"

For AST analyses, the licensee assumed a specific activity limit of 1.0 microcuries per gram (µCi/gm) dose equivalent (DE) I-131 for the RCS and limited an iodine spike to 60 µCi/gm. As such, the licensee proposed to reduce the allowable equilibrium specific activity in TS 3.4.12 and Surveillance Requirement (SR) 3.4.12.2 for the RCS from 3.5 µCi/gm to 1.0 µCi/gm DE I-131 and add a new "OR" condition to Action B to limit any iodine spike to 60 µCi/gm DE I-131. Specifically, the proposed TS changes are as follows:

- Current Limiting Condition for Operation (LCO) 3.4.12.a states, "≤ 3.5 µCi/gm DOSE EQUIVALENT I-131; and." Revised LCO 3.4.12.a would state, "≤ 1.0 µCi/gm DOSE EQUIVALENT I-131; and."

- Condition B in the action statement of TS 3.4.12 currently states, "Required Action and associated Completion Time not met." Revised Condition B would state,

  Required Action and associated Completion Time not met.
  <u>OR</u>
  DOSE EQUIVALENT I-131 > 60 µCi/gm.

- Current SR 3.4.12.2 states, "Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 3.5 µCi/gm." Revised SR 3.4.12.2 would state, "Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 1.0 µCi/gm."

The AST reanalyses has determined that the proposed new TS values provide acceptable radiological consequences of DBAs. These changes are also conservative as they provide a more restrictive limit for the specific activity in the RCS and an additional limitation that did not exist previously. Based on the above, the NRC staff concludes these TS changes are acceptable. In addition, these changes are consistent with the NUREG-1430 and TSTF-490, "Deletion of E Bar Definition and Revision to RCS Specific Activity Technical Specification."

3.2.8.2 TS 3.7.4, "Secondary Specific Activity"

For AST analyses, the licensee assumed a specific activity limit of 0.1 µCi/gm DE I-131 for the secondary coolant. As such, the licensee proposed to reduce the allowable specific activity in TS 3.7.4 and SR 3.7.4.1 for the secondary coolant from 0.17 µCi/gm to 0.1 µCi/gm DE I-131. Specifically, the proposed changes are as follows:

- Current LCO 3.7.4 states, "The specific activity of the secondary coolant shall be ≤ 0.17 μCi/gm DOSE EQUIVALENT I-131." Revised LCO 3.7.4 would state, "The specific activity of the secondary coolant shall be ≤ 0.1 μCi/gm DOSE EQUIVALENT I-131."

- Current SR 3.7.4.1 states, "Verify the specific activity of the secondary coolant is ≤ 0.17 μCi/gm DOSE EQUIVALENT I-131." Revised SR 3.4.12.2 would state, "Verify the specific activity of the secondary coolant is ≤ 0.1 μCi/gm DOSE EQUIVALENT I-131."

The AST reanalyses has determined that the new proposed TS value provides acceptable radiological consequences of DBAs. In addition, this change is a conservative change in that it provides a more restrictive limit for the specific activity in the secondary coolant. Based on the above, the NRC staff concludes that this TS change is acceptable.

3.2 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 ANO-1. The staff concludes that the licensee used analysis methods and assumptions consistent with the conservative regulatory requirements and guidance identified in Section 2.0. The staff compared the doses estimated by the licensee to the applicable criteria identified in Section 2.0 and concludes with reasonable assurance that the licensee's estimates of the EAB, LPZ, and control room doses will comply with these criteria. The staff further concludes with reasonable assurance that ANO-1, as modified by this approved license amendment, will continue to provide sufficient safety margins, with adequate defense-in-depth, to address unanticipated events and to compensate for uncertainties in accident progression analysis assumptions, and input parameters. Therefore, the proposed license amendment is acceptable with respect to the radiological consequences of DBAs. In addition, the NRC staff concludes that the licensee used analysis methods and assumptions consistent with the conservative regulatory requirements and guidance identified in Section 2.0 for the review of the electrical and environmental portions of the proposed amendment.

This licensing action is considered a full implementation of the AST. With this approval, the previous accident source term in the ANO-1 design basis is superseded by the AST proposed by the licensee. The previous offsite and control room accident dose criteria expressed in terms of whole body, thyroid, and skin doses are superseded by the TEDE criteria of 10 CFR 50.67, or fractions thereof, as defined in RG 1.183. All future radiological accident analyses performed to show compliance with regulatory requirements shall address all characteristics of the AST and the TEDE criteria as defined in the ANO-1 design basis, and modified by this amendment.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding published in the *Federal Register* on December 18, 2007 (72 FR 71708). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: D. Duvigneaud
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S. Ray
M. Yoder
A. Wang

Date: October 21, 2009

Table 3.1-1
ANO-1 Control Room Atmospheric Dispersion Factors
(χ/Q Values, sec/m^3)

Release Location	0 – 2 hours	2 – 8 hours	8 – 24 hours	1 – 4 days	4 – 30 days
Containment	3.55×10^{-3}	2.49×10^{-3}	9.85×10^{-4}	8.30×10^{-4}	6.31×10^{-4}
MSSV (1)	1.90×10^{-2}	1.23×10^{-2}	5.83×10^{-3}	3.80×10^{-3}	3.10×10^{-3}
ADV (2)	4.10×10^{-3}	2.59×10^{-3}	1.12×10^{-3}	8.32×10^{-4}	5.91×10^{-4}
PRVS (3)	4.46×10^{-3}	3.05×10^{-3}	1.36×10^{-3}	8.70×10^{-4}	7.36×10^{-4}
Main Steam Pipe	3.15×10^{-3}	2.16×10^{-3}	8.90×10^{-4}	6.61×10^{-4}	5.01×10^{-4}

(1) Main steam safety valve
(2) Atmospheric dump valve
(3) Penetration room ventilation system

Table 3.1-2
ANO-1 EAB and LPZ Atmospheric Dispersion Factors
(χ/Q Values, sec/m^3)

EAB [1]	
0-2 hours	6.8×10^{-4}
LPZ [2]	
0-8 hours	1.1×10^{-4}
8-24 hours	1.1×10^{-5}
1-4 days	4.0×10^{-6}
4-30 days	1.3×10^{-6}

[1] Exclusion area boundary
[2] Low population zone

Table 3.2-1
ANO-1 Calculated Radiological Consequences TEDE [1] (rem)

Design-Basis Accident	EAB [2]	LPZ [3]	CR [4]
Loss of Coolant Accident	10.49	2.56	3.77
Fuel Handling Accident	1.40	0.25	1.00
Steam Generator Tube Rupture Accident			
pre-accident iodine spike	2.20	0.37	2.33
concurrent iodine spike	1.26	0.23	1.00
Control Rod Ejection Accident			
containment leakage	4.73	2.28	3.40
primary-secondary leakage	3.03	1.64	4.95
Main Steam Line Break			
pre-accident iodine spike	0.45	0.19	1.84
concurrent iodine spike	2.07	1.05	3.72

[1] Total effective dose equivalent
[2] Exclusion area boundary
[3] Low population zone
[4] Control room

Table 3.3-2
ANO-1 Parameters and Assumptions for the LOCA

Parameter	Value
Power level for analyses (102% of 2568 MWt)	2619.36 MWt
Core Average Fuel Burnup	41,045 MWD/MTU
Maximum Fuel Enrichment	4.1 w/o
Margin Added to ORIGEN Source Term Results	4%
Gap Release Phase	30 sec - 0.5 hrs
Early In-Vessel Release Phase	0.5 - 1.8 hrs
Gap Release Fraction	0.05 for noble gases, halogens, and alkali metals only
Early In-Vessel Release Fractions	0.95 noble gases 0.35 halogens 0.25 alkali metals 0.05 tellurium metals 0.02 strontium and barium 0.0025 noble metals 0.0005 cerium group 0.0002 lanthanides
Iodine species distribution (%)	95.00 particulate 4.85 elemental 0.15 organic
Containment Net Free Volume	1.81×10^6 ft^3
Containment Leak Rates	0.2%/day <= 24 hrs 0.1%/day > 24 hrs
Unsprayed Containment Volume	2.00×10^5 ft^3 (rounded up)
Sump Volume	54,918 ft^3
Sprayed Containment Volume	1.61×10^6 ft^3
Containment Sprayed Fractions	0.11 unsprayed 0.89 sprayed
Containment Mixing Rates	6270 cfm unsprayed to sprayed 6270 cfm sprayed to unsprayed

Table 3.3-2
ANO-1 Parameters and Assumptions for the LOCA (continued)

Parameter	Value
Spray Removal Rates	
Elemental	20 hr^{-1} during injection, 10 hr^{-1} during recirculation until DF=200, then 0; DF=200 at 3.1 hours
Organic	No removal
Particulate	2.60 hr^{-1}until DF=50 at 3.3 hours, then 0.26 hr^{-1} until DF=1000 at 16.5 hours, then 0
Spray Initiation Time (no termination)	300 sec
Natural Deposition in Unsprayed Region	No credit taken
Amount of Containment Leakage into Penetration Rooms	50%
Penetration Room Ventilation System Filter Efficiency	99% particulates 90% elemental and organic iodines 0% noble gases
Dose Conversion Factors (DCF)	Federal Guidance Report 11 CEDE and Federal Guidance Report 12 EDE
Time of control room Isolation	10 seconds
control room Unfiltered Inleakage	82 cfm
ESF Leakage Rate	782 cc/hr (4.603E-4 cfm)
Fraction of Released Iodine in Sump Solution	1.0
Iodine Species Distribution in Sump	0.97 elemental 0.03 organic
Time to Recirculation	4257 sec (1.1825 hr)
Iodine Partition Coefficient for ESF Leakage (Flashing Fraction)	Calculated - 4.58% Used in analysis - 10%
Release Filtration Assumed	None

Table 3.3-3
ANO-1 Parameters and Assumptions for the FHA

Parameter	Value
Power level for analyses (102% of 2568 MWt)	2619.36 MWt
Core Average Fuel Burnup	41,045 MWD/MTU
Peaking Factor	1.8
Number of Fuel Assemblies in Core	177
Number of Damaged Rods	82 (six rows)
Fuel Rod Pressure Limit	1500 psig
Water Level Above Damaged Fuel	23 feet minimum
Delay Before Fuel Movement	72 hrs
Gap Fractions Released	
Kr-85	0.30
1-131 (modified per NUREG/CR-5009)	0.12
Other isotopes	0.10
Iodine Form in Pool	
Elemental	99.85%
Organic	0.15%
Iodine Form Above Pool	
Elemental	70%
Organic	30%
Pool Decontamination Factors	
Elemental Iodine	286 (limited to provide overall DF = 200)
Organic Iodine and Noble Gases	1
Offsite and CR Breathing Rate (duration of event)	3.5x104 m3/s
Offsite χ/Q (duration of event)	Table 3.2-1
Control Room χ/Q (containment more limiting than fuel handling area ventilation) (duration of event)	3.55×10^{-3} s/m^3
Dose Conversion Factors (DCF)	Federal Guidance Report 11 CEDE and Federal Guidance Report 12 EDE
Time of CR Isolation	36 seconds
CR Unfiltered Inleakage	85 cfm

Table 3.3-4
ANO-1 Parameters and Assumptions for the MSLB

Parameter	Value
Power level for analyses (102% of 2568 MWt)	2619.36 MWt
Initial Primary Coolant Activity	1.0 µCi/gm DE I-131 and 72/E-bar gross activity
Activity with Pre-existing Iodine Spike	60µCi/g I-131
Initial Secondary Coolant Activity	0.1µCi/g I-131
Accident-Initiated Iodine Spike Factor	500
Accident-Initiated Iodine Spike Duration	8 hrs
Primary-to-Secondary Leak Rate	0.5 gpm/SG
Time to Begin Cooldown (operator action)	30 min
Time to Isolation of Unaffected SG (initiation of DHR)	237.8 hrs
Time to Reach 212 F/Terminate Steam Release	251.8 hrs
Faulted SG Mass	6.00×10^4 lbm
Flashing Fraction in Unaffected SG	0.2
Partition Coefficient (faulted SG and intact via flashing and vaporization)	1.0
Partition Coefficients (intact SG via steaming)	0.01 iodines 0.001 alkali metals
RCS Mass	Maximum - 2.38×10^8 gm Minimum - 2.33×10^8 gm Maximum to produce largest equilibrium appearance rate; minimum to maximize activity concentration
SG Secondary Mass	Maximum - 2.72×10^7 gm Minimum - 1.71×10^7 gm Maximum in faulted SG to maximize release; minimum in intact SG to maximize activity concentration
Iodine Form of Secondary Release Particulate Elemental Organic	 0% 97% 3%
Dose Conversion Factors (DCF)	Federal Guidance Report 11 CEDE and Federal Guidance Report 12 EDE
Time of CR Isolation	10 seconds
CR Unfiltered Inleakage	85 cfm

Table 3.3-5
ANO-1 Parameters and Assumptions for the SGTR

Parameter	Value
Power level for analyses (102% of 2568 MWt)	2619.36 MWt
Initial Primary Coolant Activity	1.0 µCi/gm DE I-131 and 72/E-bar gross activity (Table 1.7.2-1)
Activity with Pre-existing Iodine Spike	60µCi/g I-131
Initial Secondary Coolant Activity	0.1µCi/g I-131
Accident-Initiated Iodine Spike Factor	335
Accident-Initiated Iodine Spike Duration	8 hrs
Initial Ruptured SG Tube Leak Rate	435 gpm
Primary-to-Secondary Leak Rate	150 gpd per SG
Time to Reactor Trip (full steaming until trip)	11 min
Time to Isolation of Faulted SG	34 min
Time to Isolation of Intact SG (initiation of DHR)	237.8 hrs
Flashing Fraction in Faulted SG	0.15
Partition Coefficients prior to Reactor Trip (release via condenser)	0.0001 iodines and alkali metals
Partition Coefficient after Reactor Trip (flashing and vaporization via MSSV or ADV)	1.0
Partition Coefficients after Reactor Trip (SG steaming via MSSV or ADV)	0.01 iodines 0.001 alkali metals
RCS Mass	Maximum - 2.38×10^8 gm Minimum - 2.33×10^8 gm Maximum to produce largest equilibrium appearance rate; minimum to maximize activity concentration
SG Secondary Mass	1.71×10^7 gm Minimum used to maximize activity concentration
Dose Conversion Factors (DCF)	Federal Guidance Report 11 CEDE and Federal Guidance Report 12 EDE
Time of CR Isolation	11 minutes
CR Unfiltered Inleakage	85 cfm

Table 3.3-6
ANO-1 Parameters and Assumptions for the CREA

Parameter	Value
Power level for analyses (102% of 2568 MWt)	2619.36 MWt
Core Average Fuel Burnup	41,045 MWD/MTU
Fuel Failure (rods in DNB)	14%
Peaking Factor	1.8
Fission Product Gap Fractions (RG 1.183, Appendix H, Section 1)	0.10 noble gases and iodines 0.12 alkali metals
Containment Release Iodine Species Distribution	95% particulate 4.85% elemental 0.15% organic
Secondary Release Iodine Species Distribution	0% particulate 97% elemental 3% organic
Primary-to-Secondary (P-S) Leak Rate (secondary release model)	300 gpd
Duration of Secondary Release Event (switch to DHR system)	38.25 hrs
Flashing and Vaporizing Fraction of P-S Leakage during Cooldown (no partitioning)	0.15
Containment Net Free Volume	1.81×10^6 ft^3
Containment Leak Rates	0.2%/day = 24 hrs 0.1%/day > 24 hrs
Sedimentation Coefficient (Particulates only)	0.1/hr until DF = 1000, then 0
Containment Spray	No credit taken
Penetration Room Ventilation System	No credit taken
Partition Coefficients of P-S Leakage Mixed with Secondary Liquid Inventory	0.01 iodines 0.001 alkali metals
Steam Release Rates from Secondary	2.5815×10^6 g/min 0-2 hrs 5.6977×10^5 g/min 2-38.25 hrs
RCS Mass	2.332×10^8 gm Minimum used to maximize activity concentration
SG Secondary Mass	3.411×10^7 gm Minimum for 2 SGs used to maximize activity concentration
Time of CR Isolation	10 seconds
CR Unfiltered Inleakage	82 cfm

October 21, 2009

Vice President, Operations
Arkansas Nuclear One
Entergy Operations, Inc.
1448 S.R. 333
Russellville, AR 72802

SUBJECT: ARKANSAS NUCLEAR ONE, UNIT NO. 1 - ISSUANCE OF AMENDMENT RE: USE OF ALTERNATE SOURCE TERM (TAC NO. MD7178)

Dear Sir or Madam:

The Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 238 to Renewed Facility Operating License No. DPR-51 for Arkansas Nuclear One, Unit No. 1 (ANO-1). The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated October 22, 2007, as supplemented by letters dated April 3, August 14, and September 18, 2008, and August 31, 2009.

The amendment modifies requirements of TS 3.4.12, "RCS [reactor coolant system] Specific Activity," and TS 3.7.4, "Secondary Specific Activity," as related to the use of an alternate source term (AST) associated with accident offsite and control room dose consequences. Implementation of the AST supports adoption of the control room envelope habitability controls in accordance with NRC-approved TS Task Force (TSTF) Standard Technical Specification change traveler TSTF-448, Revision 3, "Control Room Habitability."

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

Sincerely,

/RA by Alan Wang for/

N. Kaly Kalyanam, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-313

Enclosures:
1. Amendment No. 238 to DPR-51
2. Safety Evaluation

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