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

September 30, 2016

Mr. C. R. Pierce
Regulatory Affairs Director
Southern Nuclear Operating Company, Inc.
P.O. Box 1295, Bin 038
Birmingham, AL 35201-1295

SUBJECT: EDWIN I. HATCH NUCLEAR PLANT, UNIT NOS. 1 AND 2 – ISSUANCE OF AMENDMENTS REGARDING SECONDARY CONTAINMENT DRAWDOWN TIME (CAC NOS. MF6985 AND MF6986)

Dear Mr. Pierce:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 280 to Renewed Facility Operating License No. DPR-57 and Amendment No. 224 to Renewed Facility Operating License No. NPF-5 for the Edwin I. Hatch Nuclear Plant, Unit Nos 1 and 2, respectively. The amendments consist of changes to the Technical Specifications in response to your application dated October 15, 2015, as supplemented by letters dated March 16, May 9, and May 16, 2016.

The amendments revise Surveillance Requirement 3.6.4.1.3. The change increases the allowable time for the standby gas treatment system to draw down the secondary containment to negative pressure from 2 minutes to 10 minutes.

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

Sincerely,

Michael D. Orenak

Michael D. Orenak, Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-321 and 50-366

Enclosures:
1. Amendment No. 280 to DPR-57
2. Amendment No. 224 to NPF-5
3. Safety Evaluation

cc w/enclosures: Distribution via Listserv

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

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA

CITY OF DALTON, GEORGIA

DOCKET NO. 50-321

EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 280
Renewed License No. DPR-57

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

    A. The application for amendment to the Edwin I. Hatch Nuclear Plant, Unit No. 1 (the facility) Renewed Facility Operating License No. DPR-57 filed by Southern Nuclear Operating Company, Inc. (the licensee), acting for itself, Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and the City of Dalton, Georgia (the owners), dated October 15, 2015, as supplemented by letters dated March 16, May 9, and May 16, 2016, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as 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 set forth in 10 CFR Chapter I;

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

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

2. Accordingly, the license is hereby amended by page 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-57 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications (Appendix A) and the Environmental Protection Plan (Appendix B), as revised through Amendment No. 280, are hereby incorporated in the renewed license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley

Michael T. Markley, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Renewed Facility Operating License No. DPR-57 and Technical Specifications

Date of Issuance: September 30, 2016

ATTACHMENT TO LICENSE AMENDMENT NO. 280

EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 1

RENEWED FACILITY OPERATING LICENSE NO. DPR-57

DOCKET NO. 50-321

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

Remove Pages	Insert Pages
License	License
4	4
TSs	TSs
3.6-35	3.6-35

for sample analysis or instrumentation calibration, or associated with radioactive apparatus or components;

(6) Southern Nuclear, 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, Section 50.54 of Part 50, and Section 70.32 of Part 70; all applicable provisions of the Act and the rules, regulations, and orders of the Commission now or hereafter in effect; and the additional conditions specified or incorporated below:

(1) Maximum Power Level

Southern Nuclear is authorized to operate the facility at steady state reactor core power levels not in excess of 2804 megawatts thermal.

(2) Technical Specifications

The Technical Specifications (Appendix A) and the Environmental Plan (Appendix B), as revised through Amendment No. 280 are hereby incorporated in the renewed license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

The Surveillance Requirement (SR) contained in the Technical Specifications and listed below, is not required to be performed immediately upon implementation of Amendment No. 195. The SR listed below shall be successfully demonstrated before the time and condition specified:

SR 3.8.1.18 shall be successfully demonstrated at its next regularly scheduled performance.

(3) Fire Protection

Southern Nuclear shall implement and maintain in effect all provisions of the fire protection program, which is referenced in the Updated Final Safety Analysis Report for the facility, as contained in the updated Fire Hazards Analysis and Fire Protection Program for the Edwin I. Hatch Nuclear Plant, Units 1 and 2, which was originally submitted by letter dated July 22, 1986. Southern Nuclear may make changes to the fire protection program without prior Commission approval only if the changes

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. (continued)	C.2 Suspend CORE ALTERATIONS. AND C.3 Initiate action to suspend OPDRVs.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.1.1	Verify all secondary containment equipment hatches are closed and sealed.	In accordance with the Surveillance Frequency Control Program
SR 3.6.4.1.2	Verify one secondary containment access door in each access opening is closed.	In accordance with the Surveillance Frequency Control Program
SR 3.6.4.1.3	-------------------NOTE------------------- The number of standby gas treatment (SGT) subsystem(s) required for this Surveillance is dependent on the secondary containment configuration, and shall be one less than the number required to meet LCO 3.6.4.3, "Standby Gas Treatment (SGT) System," for the given configuration. -- Verify required SGT subsystem(s) will draw down the secondary containment to ≥ 0.20 inch of vacuum water gauge in ≤ 10 minutes.	In accordance with the Surveillance Frequency Control Program

(continued)

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SOUTHERN NUCLEAR OPERATING COMPANY, INC.

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA

CITY OF DALTON, GEORGIA

DOCKET NO. 50-366

EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 224
Renewed License No. NPF-5

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

   A. The application for amendment to the Edwin I. Hatch Nuclear Plant, Unit No. 2 (the facility) Renewed Facility Operating License No. NPF-5 filed by Southern Nuclear Operating Company, Inc. (the licensee), acting for itself, Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and the City of Dalton, Georgia (the owners), dated October 15, 2015, as supplemented by letters dated March 16, May 9, and May 16, 2016, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as 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 set forth in 10 CFR Chapter I;

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

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

2. Accordingly, the license is hereby amended by page 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. NPF-5 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications (Appendix A) and the Environmental Protection Plan (Appendix B), as revised through Amendment No. 224, are hereby incorporated in the renewed license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley

Michael T. Markley, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Renewed Facility Operating License No. NPF-5 and Technical Specifications

Date of Issuance: September 30, 2016

ATTACHMENT TO LICENSE AMENDMENT NO. 224

EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 2

RENEWED FACILITY OPERATING LICENSE NO. NPF-5

DOCKET NO. 50-366

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

Remove Pages	Insert Pages
License	License
4	4
TSs	TSs
3.6-34	3.6-34

(6) Southern Nuclear, 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, Section 50.54 of Part 50, and Section 70.32 of Part 70; all applicable provisions of the Act and the rules, regulations, and orders of the Commission now or hereafter in effect; and the additional conditions[2] specified or incorporated below:

(1) Maximum Power Level

Southern Nuclear is authorized to operate the facility at steady state reactor core power levels not in excess of 2,804 megawatts thermal, in accordance with the conditions specified herein.

(2) Technical Specifications

The Technical Specifications (Appendix A) and the Environmental Protection Plan (Appendix B); as revised through Amendment No. 224 are hereby incorporated in the renewed license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Additional Conditions

The matters specified in the following conditions shall be completed to the satisfaction of the Commission within the stated time periods following the issuance of the renewed license or within the operational restrictions indicated. The removal of these conditions shall be made by an amendment to the license supported by a favorable evaluation by the Commission.

(a) Fire Protection

Southern Nuclear shall implement and maintain in effect all provisions of the fire protection program, which is referenced in the Updated Final Safety Analysis Report for the facility, as contained

[2] The original licensee authorized to possess, use, and operate the facility was Georgia Power Company (GPC). Consequently, certain historical references to GPC remain in certain license conditions.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. (continued)	C.2 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	C.3 Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.6.4.1.1	Verify all secondary containment equipment hatches are closed and sealed.	In accordance with the Surveillance Frequency Control Program
SR 3.6.4.1.2	Verify one secondary containment access door in each access opening is closed.	In accordance with the Surveillance Frequency Control Program
SR 3.6.4.1.3	-------------------NOTE------------------- The number of standby gas treatment (SGT) subsystem(s) required for this Surveillance is dependent on the secondary containment configuration, and shall be one less than the number required to meet LCO 3.6.4.3, "Standby Gas Treatment (SGT) System," for the given configuration. --- Verify required SGT subsystem(s) will draw down the secondary containment to ≥ 0.20 inch of vacuum water gauge in ≤ 10 minutes.	In accordance with the Surveillance Frequency Control Program

(continued)

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 280 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-57

AND

AMENDMENT NO. 224 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-5

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

EDWIN I. HATCH NUCLEAR PLANT, UNIT NOS. 1 AND 2

DOCKET NOS. 50-321 AND 50-366

1.0 INTRODUCTION

By application dated October 15, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15288A528), as supplemented by letters dated March 16, May 9, and May 16, 2016 (ADAMS Accession Nos. ML16076A219, ML16130A681, and ML16137A793, respectively), Southern Nuclear Operating Company, Inc. (SNC, the licensee) requested changes to the Technical Specifications (TSs) for the Edwin I. Hatch Nuclear Plant (HNP), Unit Nos. 1 and 2. The supplemental letters dated March 16, May 9, and May 16, 2016, 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 or the Commission) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on November 24, 2015 (80 FR 73240).

The proposed changes would revise Surveillance Requirement (SR) 3.6.4.1.3 to increase the allowable time from 2 minutes to 10 minutes for the standby gas treatment system to draw down the secondary containment to negative pressure.

2.0 REGULATORY EVALUATION

2.1 System Description

The secondary containment at HNP consists of the Unit No. 1 reactor building (RB) (zone 1), the Unit No. 2 RB (zone 2), and the common refueling floor (zone 3). The safety function of the secondary containment is to contain, dilute, and holdup fission products, and to provide a means for an elevated release of the building atmosphere to ensure that offsite doses from a fuel handling accident (FHA) or loss-of-coolant accident (LOCA) are within the regulatory limits stated in Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.67, "Accident source term." The secondary containment safety function is accomplished in conjunction with the standby gas treatment system (SGTS). There are four SGTS trains at HNP. The 1A and 1B trains are assigned to HNP, Unit No. 1, and the 2A and 2B trains are assigned to HNP, Unit No. 2. Each train has its own set of dampers, charcoal filter train, and controls. Each filter consists of a moisture separator; an electric heater; a pre-filter; a high efficiency particulate air (HEPA) filter; charcoal absorbers; a second HEPA filter; an axial vane for HNP, Unit No. 1; and a centrifugal fan for HNP, Unit No. 2. For the secondary containment to be considered operable, it must have adequate leak tightness to ensure that the required vacuum can be established and maintained by a single operating standby gas treatment (SGT) subsystem.

To prevent ground level exfiltration of radioactive material, while allowing the secondary containment to be designed as a conventional structure, the secondary containment requires support systems to maintain the secondary containment pressure at less than atmospheric pressure. During normal operation, non-accident systems are used to maintain the secondary containment at a negative pressure. SR 3.6.4.1.3 requires verification that the secondary containment can be drawn down to greater than or equal to 0.20 inch of vacuum water gauge in less than or equal to 120 seconds using the SGT subsystem(s). SR 3.6.4.1.4 requires verification that the secondary containment can be maintained at greater than or equal to 0.20 inch of vacuum water gauge for 1 hour for each SGT subsystem at a flow rate of less than or equal to 4,000 cubic feet per minute. Both the 120-second drawdown time and the 0.20 inches of vacuum water gauge are credited in the current HNP LOCA safety analysis. The SGTS ensures that the secondary containment pressure is less than the external atmospheric pressure for the duration of the accident.

2.2 Proposed Changes

SNC proposes to increase the secondary containment drawdown time in SR 3.6.4.1.3 from 120 seconds to 10 minutes for both HNP units. This proposed change would be allowed during Modes 1, 2, and 3; during movement of irradiated fuel assemblies in the secondary containment; during core alterations; or during operations with a potential for draining the reactor vessel. The TS Bases will also be revised, as appropriate.

2.3 Regulatory Requirements and Guidance

The regulation in 10 CFR 50.36(c)(3) provides the basis for the SR 3.6.4.1.3 performance test of the time needed for containment drawdown. Specifically, 10 CFR 50.36(c)(3) states:

> Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met.

The regulation in 10 CFR 50.67 provides the radiological consequence analysis requirements for revisions to the accident source term. Specifically, 10 CFR 50.67(b)(2) states:

> The NRC may issue the amendment only if the applicant'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 [Sievert] (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.

HNP, Unit No. 1, is licensed to the 1967 proposed 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plant Construction Permits" (ADAMS Accession No. ML043310029). The following 1967 proposed General Design Criteria (GDC) are applicable for HNP, Unit No. 1, for this amendment request.

GDC 10, "Containment (Category A)," requires that:

> Containment shall be provided. The containment structure shall be designed to sustain the initial effects of gross equipment failures, such as a large coolant boundary break, without loss of required integrity and, together with other engineered safety features as may be necessary, to retain for as long as the situation requires the functional capability to protect the public.

GDC 11, "Control Room (Category B)," requires that:

> The facility shall be provided with a control room from which actions to maintain safe operational status of the plant can be controlled. Adequate radiation protection shall be provided to permit access, even under accident conditions, to equipment in the control room or other areas as necessary to shut down and maintain safe control of the facility without radiation exposures of personnel in excess of 10 CFR 20 limits. It shall be possible to shut the reactor down and maintain it in a safe condition if access to the control room is lost due to fire or other cause.

GDC 63, "Testing of Air Cleanup Systems Components (Category A)," requires that:

> Design provisions shall be made so that active components of the air cleanup systems, such as fans and dampers, can be tested periodically for operability and required functional performance.

GDC 64, "Testing of Air Cleanup Systems (Category A)," requires that:

> A capability shall be provided for in situ periodic testing and surveillance of the air cleanup systems to ensure (a) filter bypass paths have not developed and (b) filter and trapping materials have not deteriorated beyond acceptable limits.

GDC 65, "Testing of Operational Sequence of Air Cleanup Systems (Category A)," requires that:

> A capability shall be provided to test under conditions as close to design as practical the full operational sequence that would bring the air cleanup systems into action, including the transfer to alternate power sources and the design air flow delivery capability.

HNP, Unit No. 2, is licensed to the current Appendix A to 10 CFR Part 50, "General Design Criteria for Nuclear Power Plants." The following GDC are applicable for HNP, Unit No. 2, for this amendment request.

GDC 16, "Containment design," requires that:

> Reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

GDC 19, "Control room," requires that:

> A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem [0.05 Sv] whole body, or its equivalent to any part of the body, for the duration of the accident. Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

GDC 43, "Testing of containment atmosphere cleanup systems," requires that:

> The containment atmosphere cleanup systems shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the systems such as fans, filters, dampers, pumps, and valves and (3) the operability of the systems as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the systems into operation, including operation of applicable portions of

the protection system, the transfer between normal and emergency power sources, and the operation of associated systems.

Appendix E to 10 CFR Part 50, "Emergency Planning and Preparedness for Production and Utilization Facilities," paragraph IV.E.8, requires that adequate provisions shall be made and described for emergency facilities and equipment, including a licensee onsite technical support center (TSC) and an emergency operations facility from which effective direction can be given and effective control can be exercised during an emergency.

NUREG-0737, Supplement No. 1, "Clarification of TMI [Three Mile Island] Action Plan Requirements – Requirements for Emergency Response Capability" (ADAMS Accession No. ML102560009), Section 8.2.1, Item f, states that the TSC will be provided with radiological protection and monitoring equipment necessary to assure that radiation exposure to any person working in the TSC would not exceed 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.

NUREG-0696, "Functional Criteria for Emergency Response Facilities" (ADAMS Accession No. ML051390358), Section 2.6, states that, since the TSC is to provide direct management and technical support to the control room during an accident, it shall have the same radiological habitability as the control room under accident conditions.

NUREG-0800, Revision 0, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition," Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms" (ADAMS Accession No. ML003734190), provides guidance to the NRC staff for the review of alternative source term amendment requests. NUREG-0800, Section 15.0.1, states that the NRC reviewer should evaluate the proposed change against the guidance in Regulatory Guide (RG) 1.183, Revision 0, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" (ADAMS Accession No. ML003716792). Table 1 of NUREG-0800, Section 15.0.1, contains guidelines related to the acceptable calculated TEDE at the exclusion area boundary (EAB) and the low population zone (LPZ) outer boundaries of an analyzed design-basis accident (DBA).

NUREG-0800, Revision 3, Section 6.2.3, "Secondary Containment Functional Design" (ADAMS Accession No. ML063600406), provides guidance to the NRC staff for the review of the functional capability of the secondary containment system.

RG 1.52, Revision 4, "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" (ADAMS Accession No. ML12159A013), provides an acceptable methodology for the design, inspection, and testing of air filtration and iodine adsorption units of engineered-safety-features atmosphere cleanup systems in light-water-cooled nuclear power plants.

RG 1.183 provides an acceptable methodology for analyzing the radiological consequences of several DBAs to show compliance with 10 CFR 50.67. RG 1.183 provides guidance to licensees on acceptable application of alternate source term (AST) (also known as the accident source term) submittals, including acceptable radiological analysis assumptions for use in conjunction with the accepted AST. Regulatory Position 4.4 of RG 1.183 contains the accident-specific guideline values used in this review.

License Amendment No. 256 (Unit No. 1) and License Amendment No. 200 (Unit No. 2), dated August 28, 2008 (ADAMS Accession No. ML081770075), "Edwin I. Hatch Nuclear Plant, Unit Nos. 1 and 2, Issuance of Amendments Regarding Alternate Source Term (TAC Nos. MD2934 and MD2935)," used an AST methodology for analyzing the radiological consequences of four DBAs using RG 1.183. The NRC staff also considered relevant information in Chapter 15 of the HNP Updated Final Safety Analysis Report (UFSAR), which describes the DBAs and the evaluation of their radiological consequences.

3.0 TECHNICAL EVALUATION

3.1 Containment and Ventilation

The NRC staff reviewed the amendment request for any effects on the HNP, Unit Nos. 1 and 2, containment and ventilation systems. The licensee stated that no physical modifications are proposed for any systems, structures, or components (SSCs) designed for the prevention of previously analyzed events, nor does the submittal request a change in the operation or maintenance of any of those SSCs. The licensee stated that there will be no difference in the configuration of the secondary containment for a surveillance test performed under the current or proposed acceptance criterion of SR 3.6.4.1.3. Additionally, the licensee stated that the only reason for the requested change to SR 3.6.4.1.3 is to increase its operating margin for compliance. Although no modifications were proposed to be made to the SSCs, SSC maintenance, or SSC testing, the NRC staff still reviewed the HNP containment and ventilation systems against the requirements and NRC guidance for this amendment request.

The secondary containment depressurization and filtration systems should meet the guidelines of RG 1.52 and be capable of maintaining a uniform negative pressure throughout the secondary containment, as well as other areas served by the systems. The NRC staff questioned the locations of the pressure measurement devices to ensure that the proper pressure is achieved throughout the containment drawdown. In the March 16, 2016, supplemental letter, the licensee stated that a differential pressure indicator is permanently mounted in each zone of the secondary containment to measure the differential pressure of each zone. These differential pressure indicators have a reference leg to the outside, which ensures that a differential pressure of ≥ 0.20 inches of water is maintained throughout all of the secondary containment zones. The NRC staff reviewed this information and noted that a separate differential pressure indicator in each zone of the secondary containment results in the appropriate pressure measurement and permits a uniform negative pressure during containment drawdown during testing and accident conditions.

NUREG-0800, Section 6.2.3, Acceptance Criterion 3.B, states that the negative pressure differential in the secondary containment should be no less than 0.063 kilopascals (kPa) (0.25 inches water gauge) under all wind conditions to assure site boundary exposures less than those calculated for the DBA, even if exfiltration occurs. On September 11, 1995, the NRC issued License Amendment No. 198 (Unit No. 1) and License Amendment No. 139 (Unit No. 2) (ADAMS Accession No. ML013020068), which reduced the SR 3.6.4.1.3 and SR 3.6.4.1.4 acceptance criteria from greater than or equal to 0.25 inches of water gauge to greater than or equal to 0.20 inches of water gauge negative pressure. In the March 16, 2016, supplemental letter, the licensee stated that at > 0.20 inches of vacuum, there would be no exfiltration from the secondary containment at wind speeds less than 31 miles per hour (mph). Wind speeds of greater than 24 mph are not frequent, based on HNP meteorological conditions. The licensee further clarified in the May 9, 2016, supplemental letter that if the secondary containment is maintained at -0.125 inches water gauge, corresponding to the 24 mph wind, there will be no exfiltration. The NRC staff reviewed this supplemental information and determined that the negative differential pressure maintained in the secondary containment and other contiguous plant areas will be no less than 0.0504 kPa (0.20 inches water gauge) compared to adjacent regions under all wind conditions. The NRC staff also noted that even up to the infrequent wind speed of 24 mph, exfiltration will not occur when a negative pressure differential of more than 0.0315 kPa (0.125 inches water gauge) can be maintained in the secondary containment and other contiguous plant areas.

GDC 65 for HNP, Unit No. 1, and GDC 43 for HNP, Unit No. 2, state that the secondary containment drawdown system must be able to be tested, to the extent practical, to ensure that it will be able to perform its function during a LOCA. During the testing performed for SR 3.6.4.1.3, the secondary containment atmosphere is at normal operating temperature and humidity; however, the secondary containment would face significantly elevated temperatures and humidity levels during a LOCA. The licensee addressed this difference in the March 16, 2016, supplemental letter by stating that TS 5.5.7, "Ventilation Filter Testing Program (VFTP)," provides the testing to ensure that the SGTS is effective in filtering and adsorbing radioactive materials prior to their release from the secondary containment. The licensee also provided acceptable test results for a charcoal adsorber sample at 95 percent humidity, and stated that the filter efficiencies have been accounted for in the accident analyses. The NRC staff reviewed the supplemental information and noted that the licensee used appropriate methods to account for the environmental differences between the surveillance test and a LOCA. Additionally, the NRC staff determined that the licensee appropriately accounted for filter efficiencies in its analysis.

Based on the above, the NRC staff concludes that the licensee adequately addressed the impact of the proposed SR changes on the containment and ventilation system of HNP, Unit Nos. 1 and 2. Therefore, the NRC staff concludes that the requirements of 10 CFR 50.36 and the 1967 GDC 63, 64, and 65, and GDC 43 continue to be met.

3.2 Radiological Consequence Dose Analysis

HNP, Unit Nos. 1 and 2, were approved for AST methodology and the radiological dose consequence analyses for DBAs in License Amendment Nos. 256 and 200, respectively. The NRC staff evaluated the impact of modifying the HNP TSs to increase the secondary containment drawdown time from 120 seconds to 10 minutes on the design-basis radiological dose consequence analyses to ensure that the modification would not result in an increase in the main control room (MCR), TSC, EAB, or LPZ dose consequences above that stated in the criteria specified in 10 CFR 50.67 and the accident-specific design criteria outlined in RG 1.183.

In the amendment request, the licensee stated that among the four DBAs analyzed (i.e., control rod drop accident, LOCA, FHA, and main steam line break (MSLB)) at HNP, the limiting accident with respect to drawdown time of the secondary containment is the LOCA, and the other DBAs do not credit hold-up in the secondary containment; therefore, they are not impacted by the change in the secondary containment drawdown time.

3.2.1 Fuel Handling Accident

The HNP current licensing basis analyzed two cases for the FHA:

- Case 1: The licensee assumed the 120-second drawdown time for the secondary containment. Prior to that time, the licensee assumed that airborne activity is released, unfiltered, and at ground level. After secondary containment drawdown, all of the airborne activity was assumed to be collected by the SGTS and released. The release is elevated and filtered at a 95 percent efficiency for particulates and all forms of iodine.

- Case 2: The licensee took no credit for secondary containment isolation or operation of the SGTS. The release was assumed to be released, unfiltered, and at ground level for the duration of the accident.

In License Amendment Nos. 256 and 200, the NRC staff found that both FHA cases were modeled conservatively, the dose consequences for both cases were determined to meet applicable acceptance criteria, and the licensee's models were acceptable.

The NRC staff reviewed the impact of SNC's request to increase the secondary containment drawdown time in SR 3.6.4.1.3 from 120 seconds to 10 minutes on the previously analyzed FHA models. In the amendment request, SNC did not propose any changes to the current licensing basis source term, inputs, assumptions, or methodology for the FHA. Case 2 results in higher offsite and onsite radiological doses than Case 1 and, therefore, will bound Case 1. The proposed increase in drawdown time of the secondary containment has no impact on Case 2, or its resultant offsite and onsite radiological doses, because the release is assumed to be unfiltered and at ground level. The NRC staff finds that assuming no SGTS operation would result in more severe consequences than assuming SGTS operation with any finite drawdown time. Therefore, the NRC staff concludes that Case 2 bounds Case 1, and that there is no need to reevaluate the radiological doses of the FHA.

3.2.2 Main Steam Line Break (MSLB)

As stated in License Amendment Nos. 256 and 200, the HNP current licensing basis MSLB accident is defined as an instantaneous circumferential break of one main steam line outside the secondary containment, downstream of the outside isolation valve. The radiological consequences of an MSLB outside secondary containment is more severe than those that result from a break inside secondary containment. The radiological consequences of an MSLB outside secondary containment will bound the consequences of a break inside secondary containment. Thus, the NRC staff concludes that it was acceptable for SNC to only consider an MSLB outside of containment with regard to radiological consequences.

The NRC staff reviewed the impact of SNC's request to increase the secondary containment drawdown time in SR 3.6.4.1.3 from 120 seconds to 10 minutes on the previously analyzed MSLB accident. In the amendment request, SNC did not propose any changes to the current licensing basis source term, inputs, assumptions, or methodology for the MSLB accident. The proposed increase in drawdown time of the secondary containment has no impact on the MSLB outside of secondary containment, or its resultant offsite and onsite radiological doses. The NRC staff concludes that any hold-up in the secondary containment will yield lower radiological doses than a break outside of containment, and that, therefore, there is no need to reevaluate the radiological doses of the MSLB accident.

3.2.3 Control Rod Drop Accident (CRDA)

As stated in License Amendment Nos. 256 and 200, the HNP current licensing basis CRDA is defined as the rapid removal of the highest worth control blade from the reactor core due to a decoupling of the control rod drive mechanism from a cruciform control blade, resulting in a reactivity excursion. The licensee assumes that 1,189 fuel rods experience cladding failure, and 11 fuel rods experience melt. The activity released from the fuel instantaneously mixes in the reactor coolant within the pressure vessel and is released via the turbine/condenser to either the turbine building or the environment.

The NRC staff reviewed the impact of SNC's request to increase the secondary containment drawdown time in SR 3.6.4.1.3 from 120 seconds to 10 minutes on the previously analyzed CRDA. In the amendment request, SNC did not propose any changes to the current licensing basis source term, inputs, assumptions, or methodology for the CRDA. The proposed increase in drawdown time of the secondary containment has no impact on the CRDA because it does not occur in the secondary containment and does not release any activity to the secondary containment. Therefore, the NRC staff concludes that there is no need to reevaluate the radiological doses of the CRDA.

3.2.4 Loss-of-Coolant Accident (LOCA)

The HNP current design-basis LOCA analysis is based on the AST described in NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants" (ADAMS Accession No. ML041040063) and the radiological consequence analysis is provided in HNP UFSAR Section 15.3.3, "LOCA (Radiological Consequences) (Event 32)." To support increasing the secondary containment drawdown time in SR 3.6.4.1.3 from 120 seconds to 10 minutes, the licensee reanalyzed the offsite and onsite radiological consequences of the postulated LOCA. This reanalysis was performed to demonstrate that the engineered safety features designed to mitigate the radiological consequences at HNP would remain adequate following implementation of the proposed change.

The licensee described the AST reanalysis of the LOCA in the technical evaluation submitted as part of the license amendment request. Included in the evaluation are the key assumptions, parameters, and newly calculated EAB, LPZ, MCR, and TSC doses associated with implementing the proposed change. The licensee cites RG 1.183 as providing the primary assumptions for its reanalysis of the postulated design-basis LOCA. Specifically, the NRC staff guidance for analyses of the LOCA is detailed in Appendix A of RG 1.183.

3.2.4.1 Activity Source Term

The licensee does not propose any change to the current licensing basis source term. The current licensing basis is discussed in the NRC safety evaluation for License Amendment Nos. 256 and 200 and UFSAR Section 15.3.3.

3.2.4.2 Transport Methodology and Assumptions

The licensee calculated the onsite and offsite dose consequences of the design-basis LOCA by modeling the transport of activity released from the reactor core to the environment, while accounting for appropriate activity dilution, hold-up, and removal mechanisms. The licensee does not propose changes to the current licensing basis assumptions or methodology on transport in primary containment. The current licensing basis is discussed in the NRC safety evaluation for License Amendment Nos. 256 and 200 and UFSAR Section 15.3.3.

3.2.4.2.1 Assumptions on Dual Containments

Other than the one exception stated below, the licensee does not propose changes to the current licensing basis assumptions or methodology on dual containments. The current licensing basis is discussed in the NRC safety evaluation for License Amendment Nos. 256 and 200 and UFSAR Section 15.3.3. The one exception is as follows.

The primary containment (PC) leakage pathway was originally modeled by the licensee as the leakage from the PC that occurs prior to, and after, the establishment of a sustained negative pressure in the RB at 2 minutes after the initiation of the LOCA. This 2-minute time is referred to as the secondary containment drawdown period. Because the onset of gap release is not postulated to begin until 2 minutes after the initiation of the accident, the RB was considered to be completely drawn down when the activity release began. However, the licensee proposes to change the RB drawdown period to 10 minutes. With this change, the RB will no longer have an established negative pressure, as compared to the environment, by the onset of the gap release. Excluding 2.0 percent of this leakage that is released through bypass lines, all PC leakage is conservatively diluted in 50 percent of the RB volume and is released directly to the environment at ground level. After secondary containment is completely drawn down to negative pressure at 10 minutes after LOCA initiation, PC leakage into the secondary containment will be filtered by the SGTS in the secondary containment. The SGTS filters are credited with a 95 percent removal efficiency for all forms of iodine. PC activity releases through the SGTS are assumed to be released through the main stack at the maximum TS flow rate of 4,000 cubic feet per minute (cfm) per unit. The licensee recognized that it is possible for the SGTS fans of both units to be in operation, taking suction from one unit; therefore, the licensee assumed a maximized combined release rate of 8,000 cfm from one RB. The licensee's model of this release path is reflective of the proposed change and is consistent with RG 1.183. Therefore, the NRC staff concludes this proposed change to the dual containment assumptions to be acceptable.

3.2.4.2.2 Activity Removal in Primary Containment by Natural Deposition and Containment Sprays

The licensee's dose analysis assumed that natural deposition or sedimentation of particulate activity occurs in the PC. The licensee does not propose any change to the current licensing basis assumptions or methodology on natural deposition and containment sprays. The current licensing basis is discussed in the NRC safety evaluation for License Amendment Nos. 256 and 200 and UFSAR Section 15.3.3.

3.2.4.2.3 Primary Containment Leakage Bypassing Secondary Containment

The licensee does not propose any change to the current licensing basis assumptions or methodology on PC leakage bypassing secondary containment. The current licensing basis is discussed in the NRC safety evaluation for License Amendment Nos. 256 and 200 and UFSAR Section 15.3.3.

3.2.4.2.4 Main Steam Isolation Valve Leakage Pathway

Other than the one exception stated below, the licensee does not propose any change to the current licensing basis assumptions or methodology on main steam isolation valve leakage. The current licensing basis is discussed in the NRC safety evaluation for License Amendment Nos. 256 and 200 and UFSAR Section 15.3.3. The one exception is as follows.

While reviewing the LOCA AST, the licensee discovered a discrepancy in the assumed low pressure turbine/condenser free volume. Based on a Revision 0 of an SNC calculation, the value of 172,000 cubic feet had been assumed in the current licensing basis LOCA AST dose analysis. In a subsequent revision of the calculation, the volume was changed to 107,000 cubic feet. To resolve the discrepancy between the two values, the low pressure turbine/condenser free volume was recalculated as 68,026 cubic feet based on the condenser drawings. Therefore, this amendment request uses the new low pressure turbine/condenser free volume value in the LOCA dose analysis. The NRC staff concludes the new low pressure turbine/condenser free volume value to be reflective of the HNP plant design and, therefore, the use of the new value is acceptable.

3.2.4.2.5 Impaction in the Main Steam Lines (MSLs)

The licensee does not propose any change to the current licensing basis assumptions or methodology on impaction in the MSLs. The current licensing basis is discussed in the NRC safety evaluation for License Amendment Nos. 256 and 200 and UFSAR Section 15.3.3.

3.2.4.2.6 Sedimentation in Main Steam Lines and Condenser

The licensee does not propose any change to the current licensing basis assumptions or methodology on sedimentation in MSLs and condenser. The current licensing basis is discussed in the NRC safety evaluation for License Amendment Nos. 256 and 200 and UFSAR Section 15.3.3.

3.2.4.2.7 Engineered Safety Features Leakage

The licensee does not propose any change to the current licensing basis assumptions or methodology on engineered safety features leakage. The current licensing basis is discussed in the NRC safety evaluation for License Amendment Nos. 256 and 200 and UFSAR Section 15.3.3.

3.2.4.2.8 Other Bypass Leakage Pathways

Other than the one exception stated below, the licensee does not propose any change to the current licensing basis assumptions or methodology on other bypass leakage pathways. The current licensing basis is discussed in the NRC safety evaluation for License Amendment Nos. 256 and 200 and UFSAR Section 15.3.3. The one exception is as follows.

The radiological dose consequence calculated from these pathways is affected by the proposed change to the low pressure turbine/condenser free volume that is described in Section 3.2.4.2.4 of this safety evaluation. Although the activity calculated to be released via this pathway has not increased from the current licensing basis, the correction to the condenser volume causes the radiological dose consequence to increase. With the leakage rate from the condenser to the turbine building (TB) unchanged, the dose rates in the TB increase, which causes an increase in the TB air cloud shine dose to the MCR, and an increase in the dose from ingress/egress through the TB to the MCR. In addition, this proposed change adds to the radiological doses at the EAB, LPZ, and TSC. The resulting radiological doses are discussed below and presented in Tables 2, 3, and 4.

3.2.4.2.8.1 Ingress/Egress to the Turbine Building

In the May 16, 2016, supplemental letter, the licensee submitted an analysis summary of the radiation dose received by the MCR operators upon ingress/egress from the site boundary to the TB that accounted for the direct radiation shine from the RB, direct radiation shine from the turbine building, activity from the RB and TB released at ground level, and activity from the RB released through the plant stack. The licensee divided the ingress/egress into three legs: (1) site boundary to the parking lot in which the drive time is estimated to be 5 minutes, (2) parking lot to the security building in which the walking time is estimated to be 6 minutes, and (3) security building to the service building annex in which the walking time is estimated to be 4 minutes. The licensee assumed that the control room operators would exit the plant at 24 hours post-LOCA and thereafter make one roundtrip daily for ingress/egress between the site boundary and the TB. The wall thickness of the RB and TB vary; however, the licensee conservatively assumed the wall thickness to be the minimum of 1 foot 6 inches of uniform concrete. The minimum distance from the RB to the trip legs ranges from 300 feet (security building) to 1,200 feet (parking lot). The minimum distance from the TB ranges from 100 feet (security building) to 1,000 feet (parking lot). The licensee calculates the time dependent shine dose rate for each leg of the trip and multiplies by the transit time to yield the shine dose over the duration of the accident.

For the activity releases, the average atmospheric dispersion factors from each ground level and elevated release point to each leg of the trip are calculated for time periods of 24 to 96 and 96 to 720 hours using the ARCON96 computer program. These dispersion factors are then used with the activity released from the RB and TB to the environment during each time period and trip transit times to calculate inhalation and immersion doses during each leg of the trip. The operator breathing rate is assumed to be at the maximum value of 3.5×10^{-4} cubic meters per second for the entire duration of the accident in calculating the inhalation doses. The resulting doses calculated by the licensee are as follows.

Trip Leg	Site Ingress/Egress Dose (rem in TEDE)				
	Reactor Building Shine	Turbine Building Shine	Ground Level Release	Elevated Release	Total
Site boundary to parking lot	3.79E-4	1.38E-5	1.70E-3	6.17E-4	2.71E-3
Parking lot to security building	3.22E-3	1.31E-4	1.98E-2	6.05E-4	2.38E-2
Security building to service building annex	3.40E-3	1.42E-4	4.07E-2	3.61E-4	4.46E-2
Total	7.01E-3	2.86E-4	6.22E-2	1.58E-3	7.11E-2

The licensee stated that these doses are bounded by conservatisms in the current AST ingress/egress dose contribution from the TB to the MCR. The licensee further stated:

> The previously calculated control operator doses are as follows:
>
> - Control room air – 3.58 rem TEDE
> - Ingress/egress through turbine building – 1.30 rem TEDE
> - Turbine building shine into control room – 0.0059 rem TEDE
> - Other external sources – 0.03 rem TEDE
> - Total – 4.9 rem TEDE
>
> Based on operator walking speed and distance, the transit time for ingress/egress through the turbine building was estimated to be 1.2 minutes per trip. For conservatism, this was rounded up to 2 minutes in calculating the turbine building ingress/egress dose of 1.30 rem TEDE. The dose based on a transit time of 1.2 minutes would be as follows:
>
> (1.30 rem TEDE)(1.2/2.0) = 0.78 rem TEDE
>
> Hence, the conservatism in the previous turbine building ingress/egress dose was 0.52 rem TEDE (1.30- 0.78). Compared to this, the site ingress/egress dose of 0.0711 rem TEDE is negligible and the previously calculated control room dose of 4.9 rem remains bounding.

The NRC staff reviewed the assumptions, inputs, and methodology used by the licensee to assess the calculated dose from ingress/egress to the TB. The NRC staff concludes that the licensee used conservative analysis assumptions and inputs and that the current calculated dose from ingress/egress through the TB has enough margin to bound the ingress/egress dose from the site boundary to the TB.

3.2.4.2.9 Other External Shine to the Control Room

The licensee does not propose any change to the current licensing basis assumptions or methodology on other external shine to the MCR. The current licensing basis is discussed in the NRC safety evaluation for License Amendment Nos. 256 and 200 and UFSAR Section 15.3.3.

3.2.5 Technical Support Center Dose Consequence Assessment

During normal operations, the TSC ventilation system consists of an air handling unit with a condensing unit and an outside air inlet damper. During accident conditions, the outside air inlet damper will close, and the filter train air inlet damper will open, diverting the outside air through a filter train consisting of a pre-filter, an electric heater, a set of two HEPA filters, six charcoal adsorbers, and a fan unit. A surveillance procedure is performed by the licensee once every 2 years, which verifies that the TSC and the TSC mechanical room are at positive pressure with respect to the environment, and includes verification that the outside air supply to the TSC is 500 cfm. A second surveillance procedure is performed by the licensee once every 2 years that verifies that the TSC filter train is capable of providing filtered air for the pressurization of the TSC. In addition, it ensures that the outside air damper is closed. Unfiltered in-leakage can occur via the outside air inlet damper to the air handling unit if the damper's seal were to fail. The licensee performed an engineering review of the damper's failure and determined that the maximum leakage past this damper would be 130 cfm.

The current licensing basis assumes the TSC unfiltered in-leakage rate to be 10,000 cfm, the TSC filtered intake to be 500 cfm, and the TSC filter efficiency to be 90 percent for elemental iodine, organic iodine, and all particulate forms of radionuclide activity. The licensee does not propose any change to the current licensing basis methodology.

In the license amendment request, SNC proposes to increase the secondary containment drawdown time in SR 3.6.4.1.3 from 120 seconds to 10 minutes. As a result of increasing the secondary containment drawdown time, the secondary containment no longer limits the ground level release of airborne radioactive materials and no longer provides a means via SGTS for an elevated release of the building atmosphere for 10 minutes post-LOCA. The added 8 minutes to the secondary containment drawdown time causes the ground level activity concentration to increase; therefore, the offsite and onsite doses increase. As the TSC draws in outside air, the greater activity concentration in the environment leads to an increase in TSC dose. In the current licensing basis, SNC assumed a conservatively high in-leakage rate of 10,000 cfm for the TSC. To support increasing the secondary containment drawdown time from 120 seconds to 10 minutes, the licensee found it necessary to reduce some of the conservatism in the assumed TSC in-leakage rate and TSC filter efficiencies. The licensee proposes to reduce the assumed TSC in-leakage from 10,000 cfm to 1,000 cfm and to increase the assumed TSC filter efficiencies from 90 percent to 95 percent. The licensee stated that the proposed TSC unfiltered in-leakage rate of 1,000 cfm is conservative and equal to double the filtered intake flow rate, and that the proposed TSC filter efficiency of 95 percent is conservative with respect to design and surveillance test efficiencies, which are 99.97 percent and 99.95 percent, respectively. In the licensee's revised analysis, the combination of the proposed changes causes the calculated TSC dose to decrease from 3.9 rem TEDE to 3.1 rem TEDE, as presented in Table 2 below.

The NRC staff finds that the assumptions proposed by the licensee are reasonable and remain conservative and that the results of the licensee's re-analyses show that the TSC dose is less than the dose for the TSC provided in NUREG-0696, NUREG-0737, and paragraph IV.E.8 of Appendix E to 10 CFR Part 50. Therefore, the NRC staff concludes the licensee's proposed changes to the TSC dose consequences to be acceptable.

3.2.6 Control Room Habitability and Modeling

The HNP MCR is common to both units and is located in the TB. The current HNP DBA LOCA analysis, as described in UFSAR Chapter 15, credits the main control room environmental control (MCREC) system. The current licensing basis assumes an MCREC system intake flow rate of 250 cfm, recirculation flow of 2,100 cfm, unfiltered in-leakage of 115 cfm, and 95 percent filtration efficiency for elemental iodine, organic iodine, and all particulate forms of radionuclide activity. The licensee assumes that there is no delay in the initiation of the MCREC system, and that the associated filtration is available from the onset of activity release for LOCA. The licensee does not propose any change to the current licensing basis methodology. Other than the one exception stated below regarding unfiltered in-leakage, the licensee does not propose any change to the current licensing basis assumptions for the radiological dose to the MCR.

3.2.6.1 Unfiltered In-Leakage

SNC proposes to increase the secondary containment drawdown time in SR 3.6.4.1.3 from 120 seconds to 10 minutes. The increased secondary containment drawdown time causes the ground level activity concentration to increase; therefore, the offsite and onsite radiological doses increase. In addition, while reviewing the LOCA AST, the licensee discovered a discrepancy in the assumed low pressure turbine/condenser free volume, as discussed in Section 3.2.4.2.4 of this safety evaluation. As a result, it was necessary for the licensee to reduce the low pressure turbine/condenser free volume for the LOCA dose calculation, causing an increase in the activity concentrations in the condenser. With the leakage rate from the condenser to the TB unchanged, the dose rates in the TB increased, accounting for the increase in the MCR dose due to TB air shine. To support these changes, the licensee found it necessary to reduce some of the conservatism in the assumed MCR in-leakage rate. The licensee proposes to reduce the assumed MCR in-leakage from 115 cfm to 39 cfm. In the amendment request, the licensee states:

> Margins are applied at several levels with respect to the secondary containment safety function and to other functions intended to reduce off-site and on-site dose consequences.
>
> One is the control room unfiltered in-leakage rate, which is reduced from 115 cfm to 39 cfm for this analysis. However, results for the last MCR in-leakage test were actually far below 39 cfm. In fact, the in-leakage rate tests for the pressurization mode of the Main Control Room Environmental Control system, performed in April of 2015, indicated rates between 8 and 12 scfm (Ref. 5.7), roughly one third of the assumed in-leakage value. Therefore, although the margin was reduced, a significant amount of margin remains.

In the licensee's revised analysis, the reduced MCR in-leakage from 115 cfm to 39 cfm causes the calculated MCR dose to decrease from 4.32 to 3.58 rem TEDE. The resulting MCR dose calculated by the licensee is presented in Table 3 below.

The NRC staff finds that the assumptions proposed by the licensee are reasonable and remain conservative and that the results of the licensee's re-analysis show that the MCR dose is less than the dose for the MCR provided in Table 1 of NUREG-0800, Section 15.0.1, Regulatory

Position 4.4 of RG 1.183, and 10 CFR 50.67. Therefore, the NRC staff concludes the licensee's proposed changes to the MCR dose consequences to be acceptable.

3.2.7 Independent NRC Calculations

The licensee stated that the EAB, LPZ, TSC, and MCR doses estimated for a LOCA meet the applicable accident dose acceptance criteria. The major parameters and assumptions used by the licensee, and found to be acceptable to the NRC staff, are presented in the NRC safety evaluation for License Amendment Nos. 256 and 200, UFSAR Section 15.3.3, and Table 1 below. The results of the licensee's design-basis radiological consequence calculations are provided in Tables 2, 3, and 4 below. The NRC staff performed independent calculations of the dose consequences of the postulated LOCA releases, using the licensee's assumptions for input into the RADTRAD computer code. The NRC staff's calculations confirmed the licensee's dose results and, therefore, the NRC staff concludes that the EAB, LPZ, TSC, and MCR doses estimated by the licensee are acceptable.

3.2.8 Radiological Dose Consequences Conclusion

The NRC staff determined that the radiological consequences, first calculated by the licensee and then independently confirmed by the NRC staff, at the EAB, LPZ, TSC, and MCR, are within the dose criteria specified in 10 CFR 50.67 and the accident-specific dose acceptance criteria specified in Section 15.0.1 of NUREG-0800 and RG 1.183. The NRC staff also determined that the licensee used conservative analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this safety evaluation and with those stated in the UFSAR as design bases. Additionally, the NRC staff determined that the TSC dose consequences met the applicable criteria stated in NUREG-0696, NUREG-0737, and paragraph IV.E.8 of Appendix E to 10 CFR Part 50. Based on this information, the NRC staff finds that the licensee continues to meet the dose acceptance criteria in 10 CFR 50.67; NUREG-0800, Section 15.0.1; RG 1.183; NUREG-0696; NUREG-0737; and paragraph IV.E.8 of Appendix E to 10 CFR Part 50. Accordingly, the NRC staff concludes that the proposed changes to SR 3.6.4.1.3 resulting in the revised dose consequences are acceptable.

Table 1

Input Parameter	Current Licensing Basis	Proposed Change
Low Pressure Turbine/Condenser Free Volume	172,000 cubic feet	68,026 cubic feet
MCR Unfiltered In Leakage Rate	115 cubic feet per minute	39 cubic feet per minute
TSC Unfiltered In Leakage Rate	10,000 cubic feet per minute	1000 cubic feet per minute
TSC Filter Efficiency	90%	95%

3.3 <u>NRC Staff Conclusion</u>

The NRC staff concludes that the proposed changes to SR 3.6.4.1.3 result in acceptable dose consequences and the requirements of 10 CFR 50.36 continue to be met and is, therefore, acceptable.

Table 2

	Onsite Dose (rem TEDE)
	Technical Support Center
Air Dose	3.1
Regulatory Limit	5

Table 3

	Onsite Dose (rem TEDE)
	Main Control Room
Air Dose	3.58
Ingress/Egress Dose	1.30
TB air Dose	0.0059
Other External Shine Sources Dose	0.03
Total	4.9
Regulatory Limit	5

Table 4

	Offsite Dose (rem TEDE)	
Pathway	Exclusion Area Boundary	Low Population Zone
Ground	0.58	0.99
Elevated	0.03	0.11
Total	0.61	1.1
Regulatory Limit	25	25

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Georgia State official was notified of the proposed issuance of these amendments on August 18, 2016. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The NRC staff has determined that the amendments involve 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 amendments involve no significant hazards consideration, and there has been no public comment on such finding published in the *Federal Register* on November 24, 2015 (80 FR 73240). Accordingly, the amendments meet 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 amendments.

6.0 CONCLUSION

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

Principal Contributors: K. Bucholtz
S. Peng

Date: September 30, 2016

September 30, 2016

Mr. C. R. Pierce
Regulatory Affairs Director
Southern Nuclear Operating Company, Inc.
P.O. Box 1295, Bin 038
Birmingham, AL 35201-1295

SUBJECT: EDWIN I. HATCH NUCLEAR PLANT, UNIT NOS. 1 AND 2 – ISSUANCE OF AMENDMENTS REGARDING SECONDARY CONTAINMENT DRAWDOWN TIME (CAC NOS. MF6985 AND MF6986)

Dear Mr. Pierce:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 280 to Renewed Facility Operating License No. DPR-57 and Amendment No. 224 to Renewed Facility Operating License No. NPF-5 for the Edwin I. Hatch Nuclear Plant, Unit Nos 1 and 2, respectively. The amendments consist of changes to the Technical Specifications in response to your application dated October 15, 2015, as supplemented by letters dated March 16, May 9, and May 16, 2016.

The amendments revise Surveillance Requirement 3.6.4.1.3. The change increases the allowable time for the standby gas treatment system to draw down the secondary containment to negative pressure from 2 minutes to 10 minutes.

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

Sincerely,
/RA/
Michael D. Orenak, Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-321 and 50-366

Enclosures:
1. Amendment No. 280 to DPR-57
2. Amendment No. 224 to NPF-5
3. Safety Evaluation

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