August 28, 2008

Mr. Dennis R. Madison
Vice President - Hatch
Edwin I. Hatch Nuclear Plant
11028 Hatch Parkway North
Baxley, GA 31513

SUBJECT: EDWIN I. HATCH NUCLEAR PLANT, UNIT NOS. 1 AND 2, ISSUANCE OF AMENDMENTS REGARDING ALTERNATE SOURCE TERM (TAC NOS. MD2934 AND MD2935)

Dear Mr. Madison:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 256 to Renewed Facility Operating License DPR-57 and Amendment No. 200 to Renewed Facility Operating License NPF-5 for the Edwin I. Hatch Nuclear Plant, Units 1 and 2 (HNP), respectively. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated August 29, 2006, as supplemented November 6, November 27, 2006, January 30, June 22, July 16, August 13, October 18, December 11, 2007, January 24, February 4, February 25 (two letters, nos. 1389 and 0175), February 27, March 13, April 1, May 5, June 25, July 2, July 14, and August 14, 2008.

The amendments revise the licensing basis with a full scope implementation of an alternative source term (AST) for HNP.

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/

Robert E. Martin, Sr. 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. 256 to DPR-57
2. Amendment No. 200 to NPF-5
3. Safety Evaluation

cc w/encls: See next page

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DATE	08/25/08	08/25/08	04/24/08	08/15/08	08/19/08	08/28/08
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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. 50-321

EDWIN I. HATCH NUCLEAR PLANT, UNIT NO.1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 256
Renewed License No. DPR-57

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

   A. The application for amendment to the Edwin I. Hatch Nuclear Plant, Unit 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 City of Dalton, Georgia (the owners), dated August 29, 2006, as supplemented November 6, November 27, 2006, January 30, June 22, July 16, August 13, October 18, December 11, 2007, January 24, February 4, February 25 (two letters, nos. 1389 and 0175), February 27, March 13, April 1, May 5,June 25, July 2, July 14, and August 14, 2008, 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 contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 256, are hereby incorporated in the 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 before May 31, 2012.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Melanie C. Wong, Acting 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 the Technical Specifications

Date of Issuance: August 28, 2008

ATTACHMENT TO LICENSE AMENDMENT NO. 256

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
Page 4	Page 4
Page 6	Page 6
------	Page 7
TSs	TSs
iii	iii
1.1-2	1.1-2
3.4-12	3.4-12
3.6-12	3.6-12
3.6-29	3.6-29
3.6-30	3.6-30
3.6-31	3.6-31
3.6-32	3.6-32
3.6-33	3.6-33
3.6-34	3.6-34
3.6-35	3.6-35
3.6-36	3.6-36
3.6-37	3.6-37
3.6-38	3.6-38
3.6-39	3.6-39
3.6-40	3.6-40
-------	3.6-41
-------	3.6-42
-------	3.7-20
-------	3.7-21

Pages 3.6-31 through 3.6-42 are included for renumbering of the pages only.

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. 200
Renewed License No. NPF-5

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

   A. The application for amendment to the Edwin I. Hatch Nuclear Plant, Unit 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 City of Dalton, Georgia (the owners), dated August 29, 2006, as supplemented November 6, November 27, 2006, January 30, June 22, July 16, August 13, October 18, December 11, 2007, January 24, February 4, February 25 (two letters, nos. 1389 and 0175), February 27, March 13, April 1, May 5,June 25, July 2, July 14, and August 14, 2008, 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 contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 200, are hereby incorporated in the 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 before May 31, 2011.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Melanie C. Wong, Acting Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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

Date of Issuance: August 28, 2008

ATTACHMENT TO LICENSE AMENDMENT NO. 200

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
Page 4	Page 4
Page 6	Page 6
--------	Page 6a
TSs	TSs
iii	iii
1.1-2	1.1-2
3.4-12	3.4-12
3.6-12	3.6-12
3.6-29	3.6-29
3/6-30	3/6-30
-------	3.7-20
-------	3.7-21

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 256 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-57

AND

AMENDMENT NO. 200 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 August 29, 2006, (the License Amendment Request, LAR), as supplemented on November 6, November 27, 2006, January 30, June 22, July 16, August 13, October 18, and December 11, 2007, January 24, February 4, February 25 (two letters), February 27, March 13, April 1, May 5, June 25, July 2, July 14, and August 14, 2008, (References 1 – 18, 28, 29, 30), Southern Nuclear Operating Company, Inc. (SNC, the licensee), requested changes to the Technical Specifications (TSs) and licensing bases for the Edwin I. Hatch Nuclear Plant, Unit Nos. 1 and 2 (HNP). The licensee's supplement dated August 14, 2008, 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 original proposed no significant hazards consideration determination as published in the *Federal Register* on July 23, 2008 (73 FR 42834).

On May 25, 2006, the NRC staff issued amendments to the operating licenses for the HNP that included a condition to the operating licenses that allowed the crediting of administering potassium iodide (KI) to reduce main control room (MCR) operator thyroid dose in the event of control room inleakage during certain design basis accidents (DBAs) (Reference 26). That amendment was issued with the expectation that the licensee would submit a further LAR proposing revisions to the licensing and design basis to address this issue. SNC's August 29, 2006, application, as supplemented, responds to the issued identified in the May 25, 2006, license amendment by proposing changes to the HNP licensing basis that implement alternative source term (AST) methodology for analyzing certain DBA radiological consequence analyses. The August 29, 2006 LAR, and its supplements, as listed above, including TS changes, license conditions, licensee commitments, plant modifications and all other issues in those submittals, were reviewed by the NRC staff as one integrated license amendment application. This approved license amendment authorizes a change in the licensing basis for HNP to be implemented upon completion of all of the required actions identified in those submittals and this license amendment. That implementation must be completed by the dates specified in the license amendment: May 31, 2012 for HNP Unit 1 and May 31, 2011 for HNP Unit 2. This SE includes the following technical evaluation sections:

2.0 EVALUATION

2.1 Radiological Consequences Analyses

2.1.1 Regulatory Evaluation

This SE section is based on the licensee's submittals dated August 29, 2006, February 25, 2008, and May 5, 2008. The current HNP licensing basis uses a source term that is based on the Techinical Information Document (TID) -14844, "Calculation of Distance Factors for Power and Test Reactor Sites," to calculate the radiological consequences of postulated DBAs. The amendment request provides the TS changes and DBA radiological consequence analyses associated with a full-scope implementation of an AST, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50.67, "Accident Source Term." The licensee used the guidance described in Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" (RG 1.183). The SNC submittals contain the reanalysis and licensing basis alternative method changes necessary to meet Generic Letter 2003-01, "Control Room Habitability," objectives.

The NRC staff evaluated the licensee's analysis of the radiological consequences of postulated DBAs against the dose criteria specified in 10 CFR Part 50.67. The applicable criteria are 5 rem Total Effective Dose Equivalent (TEDE) in the control room for the duration of the event, 25 rem TEDE at the exclusion area boundary (EAB) for the worst two hours, and 25 rem TEDE at the outer boundary of the low population zone (LPZ) for the duration of the event. The dose acceptance criterion in the Technical Support Center (TSC) is accepted to be 5 rem TEDE for the duration of the accident to show consistency with NUREG-0737, "Clarification of TMI Action Plan Requirements" and Paragraph IV.E.8 of Appendix E to 10 CFR Part 50.

The regulatory requirements upon which the NRC staff based its acceptance include those in General Design Criteria (GDC) 19, "Control room," and the accident dose criteria in 10 CFR 50.67, as supplemented in Regulatory Position 4.4 and Table 6 of RG 1.183 and Standard Review Plan (SRP) 15.0.1, "Radiological Consequence Analyses Using Alternative Source Term." The licensee has not proposed any deviation or departure from the guidance provided in RG 1.183. The NRC staff review also includes reliance on following additional regulatory codes, guides, and standards, as well as the HNP Final Safety Analysis Report (FSAR) and TSs.

- NUREG-0800 SRP Section 2.3.4, "Short-Term Diffusion Estimates for Accidental Atmospheric Releases."
- NUREG-0800 SRP Section 6.4, "Control Room Habitability Systems."
- NUREG-0917, "Nuclear Regulatory Commission (NRC) Staff Computer Programs for Use with Meteorological Data."
- RG 1.23, Rev. 0, "Onsite Meteorological Programs."
- RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants."

- RG 1.197, “Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors.”

2.1.2 Technical Evaluation

The NRC staff reviewed the regulatory analysis and technical assessment performed by the licensee in support of its LAR, as they relate to the radiological consequences of DBA analyses. Information regarding these analyses was provided in LAR Enclosure 1. The NRC staff reviewed the assumptions, inputs, and methods used by the licensee to assess the impacts of the proposed license amendment. The NRC staff also performed independent calculations to confirm the conservatism of the licensee’s analyses. However, the findings of this SE are based on the descriptions and results of the licensee’s analyses and other supporting information submitted by the licensee.

2.1.2.1 Atmospheric Dispersion Estimates

The licensee previously generated control room atmospheric dispersion factors (χ/Q values) for postulated ground level and elevated releases in support of HNP Unit 1 and Unit 2 License Amendment Nos. 214 and 155, respectively, dated October 22, 1998 (Agencywide Document Access and Management System (ADAMS) Accession No. ML9811020310). However, these calculations were based upon a prior version of the ARCON computer code and a single year of meteorological data. The SE associated with License Amendment Nos. 214 and 155 recommended that the licensee use a methodology that calculates centerline χ/Q values for time periods up to eight hours in duration when making additional calculations in the future. Therefore, for the current LAR, SNC recalculated the limiting ground level and elevated release χ/Q values using ARCON96 (NUREG/CR-6331, Revision 1, "Atmospheric Relative Concentrations in Building Wakes"), which calculates centerline χ/Q values for time periods up to eight hours. In addition, SNC used three years of onsite meteorological data collected from 1996 through 1998. Other inputs and assumptions are those used to support License Amendment Nos. 214 and 155. The resulting χ/Q values for the Loss of Coolant Accident (LOCA), Main Steam Line Break (MSLB) accident, control rod drop accident (CRDA) , and Fuel Handling Accident (FHA) represent a change from the χ/Q values used in the current HNP FSAR, Chapter 15, “Safety Analysis." The licensee used previously generated χ/Q values listed in the HNP FSAR for postulated releases to the EAB and LPZ.

Meteorological Data

The licensee generated new control room χ/Q values for the HNP Units 1 and 2 ground level and elevated release dose assessments using site meteorological data collected from 1996 through 1998. The data was provided for NRC staff review in the form of hourly meteorological data files in the format suitable for input to the ARCON96 atmospheric dispersion computer code. The wind speed and wind direction data used to generate the new control room χ/Q values for both postulated ground level and elevated releases from the 120-meter stack were measured on the HNP onsite meteorological tower at heights of 10 meters and 60 meters above the ground. Temperature difference data, which were used to determine atmospheric stability class, were measured between the 60-meter and 10-meter levels. The combined data recovery of the wind speed, wind direction, and stability data was in the 90^{th} percentile at both levels during each of the 3 years. This meets the data recovery recommendation of RG 1.23, Revision 0, “Onsite Meteorological Programs.”

The NRC staff performed a quality review of the 1996 through 1998 hourly meteorological database using the methodology described in NUREG-0917, "Nuclear Regulatory Commission Staff Computer Programs for Use with Meteorological Data." Wind speed and wind direction frequency distributions for each measurement channel were reasonably similar from year to year. Wind speeds at the 10-meter level were reported to be lighter than winds that occurred in the early 1970's as summarized in the HNP FSAR and lighter than expected when compared with the 1996 through 1998, using wind data extrapolated from the 60-meter level to the 10-meter level. However, because the postulated height of effluent release for the limiting cases was at 49.7 meters, the ARCON96 computer code used the 60-meter data as its primary data source. Thus, the lighter 10-meter wind speeds did not have a significant impact on the χ/Q calculations for this LAR.

With respect to the reported atmospheric stability measurements, the daily durations of stable and unstable conditions were generally consistent with expected meteorological conditions as were the occurrences with respect to time of day. While stable and neutral conditions were generally reported to occur at night and unstable and neutral conditions during the day, the NRC staff noted an overall higher occurrence of extremely unstable and extremely stable conditions and a lower occurrence of neutral stability conditions than typically expected for measurements between the 60-meter and 10-meter levels.

In SNCs August 13, 2007 response (Reference 7) to a NRC staff request for additional information (RAI) regarding wind data, the licensee stated that some growth of trees occurred in the vicinity of the meteorological tower which may have affected the 10-meter wind and atmospheric stability measurements from 1996 through 1998. Following a walk down in 2000, trees were cut back to meet the guidance in RG 1.23, proposed Revision 1, which is the document that SNC considers its licensing basis for onsite meteorological measurements. The NRC staff notes that this draft guidance is less restrictive than that provided in RG 1.23, Revision 1, “Meteorological Monitoring Programs for Nuclear Power Plants” (March 2007). The licensee also provided a table to show that the average fifth percentile 10-meter wind speeds increased only slightly after the trees were cut back, from 0.40 meters per second (m/s) from 1996 through 1998 to 0.45 m/s from 2004 through 2006 and noted that the small increase could be due to typical climatic variations. The licensee stated that slightly lower wind speeds would result in more limiting χ/Q values. The NRC staff agrees that lower winds speeds should generally result in more limiting χ/Q values when using some atmospheric dispersion models, but notes that when using ARCON96, the limiting χ/Q values are typically associated with moderate wind speeds since ARCON96 models enhanced dispersion due to meander for light winds. Thus, when using ARCON96, an overall higher frequency of lower winds in the data does not necessarily result in more limiting χ/Q values. However, as discussed in the following section on control room dispersion factors, the NRC staff generated a sample comparison set of χ/Q values using only the 60-meter data and found that the limiting χ/Q values remained those that utilized the 60-meter wind measurements as its primary data source for the ground level release scenarios.

In the licensee’s RAI response with regard to the atmospheric stability measurements, the licensee noted that a more frequent occuring of stable conditions should result in the generation of more limiting χ/Q values than if the reported frequency occurrences were actually lower. The NRC staff agrees with this conclusion when using ARCON96 to calculate χ/Q values for postulated ground level releases. Further, an overestimate of unstable conditions should generally result in more limiting χ/Q values when making calculations for elevated releases as unstable conditions would result in the effluent being dispersed to ground level more quickly than under stable conditions.

NRC Regulatory Issue Summary 2006-04, "Experience with Implementation of Alternative Source Terms," states that when $\chi/_{Q}$ values are calculated using ARCON96 for both ground level and elevated releases, two or more files of meteorological data should be used in order to input data representative of each potential release height. The HNP stack is 120 meters in height and 259 meters from the control room intake. Therefore, in response to a NRC staff RAI to demonstrate that use of the 10-meter and 60-meter data was acceptable, by letter dated April 1, 2008 (Reference 16), the licensee also provided wind speed and direction data measured from 1996 through 1998 at heights of 10 meters and 100 meters above the ground. Temperature difference data were measured between the 100-meter and 10-meter levels. These data were not used to generate the $\chi/_{Q}$ value inputs to the dose assessment, but were used to confirm that use of the 10- and 60-meter data resulted in $\chi/_{Q}$ values that were more limiting for releases from the stack than would result from use of the 10-meter and 100-meter data. The NRC staff reviewed the meteorological data using the methodology described in NUREG-0917. The NRC staff found the data generally consistent with the measurements made at the 10-meter and 60-meter levels and acceptable for use in performing the comparison.

For the reasons cited above, the NRC staff concludes that the 1996 through 1998 meteorological data measured at the HNP site provide an acceptable basis for making atmospheric dispersion estimates for use in the dose assessments performed in support of this LAR.

Control Room Atmospheric Dispersion Factors

RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," states that ARCON96 is an acceptable methodology for assessing control room $\chi/_{Q}$ values for use in DBA radiological analyses. The NRC staff evaluated the applicability of the ARCON96 model and concludes that there are no unusual siting, building arrangements, release characterization, source-receptor configuration, meteorological regimes, or terrain conditions that preclude use of this model in support of the current LAR for HNP. When generating both ground level and elevated release $\chi/_{Q}$ values, the license continued to use the ARCON96 default values for surface roughness length and averaging sector width constant rather than the revised values listed in RG 1.194. However, results of comparison calculations performed by the NRC staff showed that any differences in the resultant $\chi/_{Q}$ values were not significant.

To generate $\chi/_{Q}$ values for elevated releases, the licensee used 1996 through 1998 meteorological data measured at the 10- and 60-meter levels because data recovery for wind speed at the 100-meter level was slightly less than the RG 1.23 recommended recovery of at least 90 percent. The NRC staff requested information to confirm that use of the 10-meter and 60-meter data was limiting. By letters dated January 24, 2008 (Reference 10), and April 1, 2008 (Reference 16), the licensee provided the results of its assessment, as well as the related inputs and assumptions. The NRC staff reviewed the information provided by the licensee, performed comparison calculations and confirmed the licensee's conclusion that use of the 10- and 60-meter data was limiting.

For elevated releases, the NRC staff notes that the licensee did not use the procedure described in Section 3.2.2, "Elevated (Stack) Releases," of RG 1.194 which combines results using the ARCON96 and PAVAN (NUREG/CR-2858, "PAVAN: An Atmospheric Dispersion Program for Evaluating Design Basis Accidental Releases of Radioactive Materials from Nuclear Power Stations") atmospheric dispersion models. The procedure is intended to provide an assessment tool when the control room intake is located close to the base of a tall stack and the elevated release model in ARCON96 generates "negligibly low" $\chi/_{Q}$ values. However, RG 1.194 states that holders of operating licenses may continue to use $\chi/_{Q}$ values determined with methodologies previously approved by the NRC staff and documented in the facility's FSAR to the extent that these values are appropriate for the application in which they are being used. Licensees may also continue to use the licensing basis methodology for re-generating the approved $\chi/_{Q}$ values using more recently collected meteorological data sets unless changes are deemed necessary to ensure adequate protection of the health and safety of the public. Because the licensee met the criteria in RG 1.194 and because the source/receptor configuration resulted in $\chi/_{Q}$ values that NRC staff did not find to be "negligibly low," the NRC staff has judged that the licensee's use of only the ARCON96 methodology is acceptable in this specific case.

For ground level releases, the licensee provided a comparison table of $\chi/_{Q}$ values generated in support of Amendment Nos. 214 and 155 to identify the limiting cases to be revised for the current LAR. The licensee also provided associated input information for postulated releases from the Unit 1 reactor building, Unit 1 and 2 reactor vents, and six turbine building release locations. The NRC staff performed comparison calculations using wind data from only the 60-meter level and the surface roughness length and averaging sector width constant values listed in RG 1.194. Initially, it appeared that a point release from the reactor building wall resulted in more limiting $\chi/_{Q}$ values than from the reactor building vent. However, in the January 24 and April 1, 2008 RAI responses, the licensee provided information to show that the release from the reactor building vent was limiting because the release from the containment building wall could be modeled as a diffuse release as provided for in RG 1.194. The NRC staff made a confirmatory evaluation of the resulting atmospheric dispersion estimates by running the ARCON96 computer model and, as a result, agrees with the licensee's conclusion that postulated releases from the plant vent result in the limiting $\chi/_{Q}$ values.

On the basis of this review, the NRC staff concludes that the $\chi/_{Q}$ values for the HNP LOCA, MSLB, CRDA and FHA releases to the control room air intake, as presented in Table 3.1.1 of this SE, are acceptable for use in the DBA control room dose assessment.

Offsite Atmospheric Dispersion Factors

The licensee used previously generated and NRC-accepted licensing basis EAB and LPZ $\chi/_{Q}$ values to assess the radiological consequences of the LOCA, MSLB, CRDA and FHA postulated in this LAR. These $\chi/_{Q}$ values are presented in Table 2.3-11 of the HNP Unit 2 FSAR. Section 2.3 "Meteorology," of the HNP Unit 1 FSAR, states that meteorological information for HNP Unit 2 also applies to HNP Unit 1 as the data are for the plant site in general. The $\chi/_{Q}$ values are also listed in Table 3.1.2 of this SE.

Radiological Consequences of Design Basis Accidents

To support the proposed implementation of an AST, the licensee analyzed the radiological dose consequences and provided all major inputs and assumptions for the DBAs of LOCA, MSIB, CRDA and FHA.

As a minimum, effort to revise the HNP licensing basis to incorporate a full implementation of the AST, RG 1.183, Position 1.2.1, specifies that the DBA LOCA must be reanalyzed using the appropriate guidance therein. In accordance with this RG 1.183 guidance, the licensee re-analyzed the four DBAs listed above, which includes the design basis LOCA at HNP.

The licensee's submittal reports the results of the radiological consequence analyses for the above DBAs to show compliance with 10 CFR 50.67 dose acceptance criteria, or fractions thereof, as defined in SRP 15.0.1 and RG 1.183, for doses offsite and in the control room. 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, "Determination of exclusion area, low population zone, and population center distance," and GDC 19. The subject LAR, as supplemented, is considered a full implementation of the AST.

RG 1.183, Regulatory Position 3.1, "Fission Product Inventory", states that "The inventory of fission products in the reactor core and available for release to the containment should be based on the maximum full power operation of the core with, as a minimum, current licensed values for fuel enrichment, fuel burnup, and an assumed core power equal to the current licensed rated thermal power times the Emergency Core Cooling System (ECCS) evaluation uncertainty. The period of irradiation should be of sufficient duration to allow the activity of dose-significant radionuclides to reach equilibrium or to reach maximum values. The core inventory should be determined using an appropriate isotope generation and depletion computer code such as ORIGEN2 or ORIGEN-ARP." For accident analyses postulating fuel damage, and in accordance with the guidance of RG 1.183, the licensee calculated the core isotopic inventory available for release using the ORIGEN2 isotope generation and depletion computer code, and then multiplied the isotopic specific activities by the relevant power level and release fractions. However, because the licensee is interested in providing margin for future fuel changes or power uprates, SNC also assumed the calculated isotopic inventory was multiplied by 10%, or a factor of 1.1. The licensee accounted for ECCS uncertainty by adding 0.5% to the maximum full power as well. This uncertainty allowance is consistent with the Measurement Uncertainty Recovery granted to HNP in License Amendment Nos. 238 and 180 on September 23, 2003 (ADAMS Accession No. ML032590944). The NRC staff finds this uncertainty allowance and the licensee's implementation of the approved isotope generation and depletion computer code to be acceptable for establishing the core inventory for AST accident analyses.

As stated in RG 1.183, the release fractions associated with the light water reactor (LWR) core inventory released into containment for the design basis LOCA and non-LOCA events have been determined to be acceptable for use with currently approved LWR fuel with a peak burnup of 62,000 megawatt days per metric ton of uranium (MWd/MTU), provided that the maximum linear heat generation rate does not exceed a 6.3 kilowatt per foot (kw/ft) peak rod average power for burnups exceeding 54,000 MWd/MTU. The licensee states that all HNP fuel conforms to these criteria, and in their February 25, 2008 response to the NRC staff RAI

(Reference No. 12), the licensee confirmed that appropriate measures will be implemented to assure that HNP will operate in compliance with the fuel burnup parameters delineated in Footnote 11 to Table 3 of RG 1.183.

To perform independent confirmatory dose calculations for the DBAs, the NRC staff used 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. The RADTRAD code, developed by the Sandia National Laboratories for the NRC, estimates transport and removal of radionuclides and the resulting radiological consequences at selected receptors.

The following sections discuss the NRC staff review of the DBA safety assessment performed by the licensee to support the LAR submittal of August 29, 2006, including supporting supplements.

2.1.2.2 Loss of Coolant Accident

The current HNP design basis LOCA analysis is based on the traditional accident source term described in TID-14844. The current licensing basis radiological consequence analysis for the postulated LOCA is provided in the HNP FSAR Section 15.3.3, "Loss-of-Coolant Accident." To support implementation of the AST, as requested by the subject LAR, the licensee reanalyzed the offsite and control room radiological consequences of the postulated LOCA. This reanalysis was performed to demonstrate that the engineered safety features (ESFs) designed to mitigate the radiological consequences at HNP will remain adequate following implementation of the AST.

The licensee described the AST-based reanalysis of the LOCA in the safety analysis submitted as part of the LAR. Included in the assessment are the key assumptions, parameters, and newly calculated offsite and control room doses associated with implementing the AST methodology. The licensee cites RG 1.183 as providing the primary assumptions for their 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.

Activity Source

For the LOCA analysis, the licensee assumed that the core isotopic inventory available for release into the containment, is based on maximum full power operation of the core at 2,818 megawatts thermal (MWth), or 1.005 times the current licensed thermal power level of 2,804 MWth, in order to account for the ECCS flow uncertainty. In addition, the licensee increased the core isotopic activities by 10% to allow for possible future power uprates. The burnup and enrichment parameters assumed when determining the core isotopic inventory are within current licensed limits for fuel at HNP.

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 1 and 4, respectively. Also consistent with RG 1.183 guidance, the licensee assumes that the speciation of radioactive iodine released from failed fuel is 95 percent (%) aerosol (particulate), 4.85% elemental, and 0.15% organic. The speciation of radioactive iodine for coolant releases, such as from ECCS, is 97% elemental and 3% organic.

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 core to the environment, while accounting for appropriate activity dilution, holdup, and removal mechanisms. The NRC staff has reviewed the licensee's assessment of the following potential post-LOCA activity release pathways:

- Primary Containment (PC) Leakage to Secondary Containment
- PC Leakage Bypassing Secondary Containment
    - Main Steam Isolation Valve (MSIV) Leakage Pathway
    - Other Bypass Leakage Pathways
- Engineered Safety Feature Leakage

Also, the NRC staff reviewed the licensee's assessment of the following potential post-LOCA shine dose pathways:

- Turbine Building Cloud Shine to the Control Room
- Ingress/Egress through the TB to the Control Room
- Other External Shine to the Control Room

Consistent with regulatory requirements, the licensee assumed a loss of offsite power (LOOP) concurrent with the design basis LOCA. Subsequently, the licensee has assumed that, as a worst case, the single failure of an emergency diesel generator (EDG) delays the startup of ECCS for 2 hours after the onset of gap release. Additionally, to conservatively limit credit for deposition of activity in piping, the licensee assumed that an inboard MSIV fails in the stuck open position creating an unrestricted flow path to the outboard MSIV, as discussed in the following section of this SE.

For releases into containment, the licensee assumed that activity released from the reactor coolant system begins to mix between the drywell and the suppression chamber (torus) airspace volumes, at 2.03 hours, approximately coincident with the hypothetically modeled restoration of ECCS, and completion of the early in-vessel activity release phase, which is postulated to be complete 122 minutes after accident initiation. Before this time, the releases are only mixed in the drywell airspace. The licensee calculated that, at the time the ECCS is restored, the thermohydraulic response of cooling water quenching the molten core and core debris in the PC will result in the mixing of the drywell and torus airspace volumes. This assumption is acceptable for the Mark I containment design of HNP, as it is configured with downcomers from the drywell that extend below the surface of the torus coolant (wetwell). The licensee calculates this mixing to occur at time-dependent rates after 2.03 hours; this mixing profile is shown in Table 3.2.1. The licensee took no credit for the activity decontamination, or scrubbing, associated with such activity releases into the wetwell.

By crediting the HNP Standby Liquid Control System (SLCS) capability to introduce sodium pentaborate to act as a buffer into the reactor coolant, the licensee has determined that the suppression pool pH remains above 7 for the duration of the accident. Therefore, in analyzing activity transport from containment, it was unnecessary for the licensee to consider re-evolution of iodine dissolved in the coolant. This analysis of post-LOCA suppression pool pH is reviewed in section 2.7 of this SE and and the reliability for the SLC system to perform this function is reviewed in section 2.5 of this SE.

The following subsections detail the NRC staff review of the licensee's analysis of the post-accident activity release paths and contributors to both control room and offsite dose, as mentioned above.

PC Leakage to Secondary Containment

The HNP current design basis containment leak rate (L_a) of 1.2 percent weight per day (% per day) at containment peak pressure, as reflected in the leak rate limit in HNP TS 5.5.12, is assumed in the AST LOCA re-analysis. The design basis leak rate of 1.2 % per day was reduced to 60% at 24, then to 50% at 72 hours and for the remaining accident duration. This reduction was acceptably justified by the analogous containment pressure and temperature reductions calculated at those same time steps. This pathway was modeled by the licensee as the leakage from the PC that occurs prior to, and after, a sustained negative pressure in the Reactor Building (RB) is established at 2 minutes after the initiation of the LOCA. This 2-minute time is referred to as the drawdown period. Therefore, because the onset of gap release was not postulated to begin until 2 minutes after the initiation of the accident, the RB was considered to be completely drawn down when activity release begins. So, excluding 2.0 % of this leakage, which is released through bypass lines, all PC leakage is conservatively diluted in 50% of the RB volume and filtered by the standby gas treatment system (SGTS) of the secondary containment. The SGTS filters are credited with a 95% removal efficiency for all forms of iodine. PC activity releases through the SGTS were assumed to be released through the plant stack at the maximum TS flow rate of 4000 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 8000 cfm from one RB. The licensee's model of this release path is acceptable to the NRC staff.

Consistent with RG 1.183 guidance, the licensee assumed that the release from the core enters the drywell only. Only after the end of the release, 122 minutes (i.e., 2.03 hours) following accident initiation, did the licensee begin to credit mixing within the entire PC (drywell plus torus airspace). As mentioned above, the licensee conservatively calculated the rate of mixing between the two primary containment volumes over the duration of the accident, and Table 3.2.1 shows the mixing profile.

Activity Removal in PC by Natural Deposition and Containment Sprays

The licensee's dose analysis assumed natural deposition, or sedimentation, of particulate activity occurs in primary containment. Rather than using a deterministic formula to calculate the removal associated with natural deposition in the containment (i.e., NUREG/CR-6189 or NUREG-0800 SRP Chapter 6.5.2), the licensee employed a proprietary code to comprehensively model this phenomenon along with the other credited activity removal phenomena due to sprays. Generally, the effect of natural deposition is largely reduced when compared to the activity removal of containment sprays. To confirm the conservatism of the licensee's model, the NRC staff used the accepted simplified natural deposition model from NUREG/CR-6189, referred to as the Powers natural deposition model, as implemented in the RADTRAD dose consequence computer code. The NRC staff accepts use of the 10^{th} percentile confidence interval (90 percent probability) natural deposition removal values implemented in the RADTRAD code, accordingly, that is used for the NRC staff confirmatory calculation to support this evaluation. The Powers natural deposition model was derived by correlation to results of Monte Carlo uncertainty analyses of detailed models of aerosol behavior in the containment under accident conditions. The NRC staff modeled the combined removal by sprays and natural deposition in primary containment and compared the resulting dose consequence to that calculated by the licensee. Also, the NRC staff assessed the individual activity removal attributable to these two mechanisms. The results confirmed that the effect of natural deposition is largely overshadowed by that of sprays, and that the licensee's model of overall PC activity removal is conservative.

While sedimentation of activity in containment was credited from the onset of gap release, credit for activity removal by drywell sprays was taken from the time sprays are manually initiated at 15 minutes into the accident and ceased at 24 hours. This timing bounds the time the licensee calculated to detect an immersion dose rate of 200,000 rem/hr in the drywell and then initiate sprays. The containment spray characteristics were conservatively modeled by the licensee as they were consistent with assuming credit for only one operating residual heat removal (RHR) pump.

PC Leakage Bypassing Secondary Containment

The licensee models this pathway as leakage through the lines that penetrate the PC and the RB. It is postulated that leakage from the PC through penetrations and the closed containment isolation valves (CIVs) in these penetrations would bypass the RB and SGTS filters, thereby resulting in an unfiltered release at ground level. The licensee assumed that all leakage bypassing secondary containment was directed into the TB volume. This activity was then assumed to be mixed in a conservatively reduced percentage of the TB free volume. After 9 hours, the licensee assumed that the exhaust capacity of one TB fan, 15,000 cfm, was available to purge the TB volume of accumulated activity. The acceptability of these assumptions and the credit taken for the TB volume and its associated exhaust ventilation system is evaluated in this SE, but particularly in section 2.5. The TB exhaust ventilation system was credited by the licensee in this same manner for analyses of the CRDA and MSLB accidents as well.

Consistent with their current licensing basis, the licensee assessed all primary containment penetrations to identify the leakage paths that do not terminate within the secondary containment and should be considered as potential secondary containment bypass leakage pathways.

MSIV Leakage Pathway

A total leakage of 100 standard cubic feet per hour (scfh), with no limit on leakage per main steam line (MSL), was assumed for MSIV leakage. The licensee has proposed a revision to the HNP TSs to reflect this new limit. The actual leak rate used for analysis is an adjusted value calculated by the licensee to account for postulated actual pressures and temperatures. The licensee's RAI response of February 25, 2008 (reference 12) presented the equation by which this standard flow rate was adjusted to accident conditions. The licensee states, and the NRC staff agrees, that volumetric flow remains constant with respect to changes in upstream pressure, as long as the gas composition and the gas temperature remain constant. Based upon this fact, the licensee calculated a volumetric flow rate, adjusted from test to accident conditions, by accounting for maximum accident temperature and steam-gas mixture (as characterized by density) postulated to be in the MSL. The licensee calculated the maximum adjusted flow rate for testing at, or above, maximum post-accident pressure to be equal to 144 scfh. The licensee adjusted the MSIV leak rate to the as-analyzed accident conditions entering the MSL from the drywell, exiting the MSL to the condenser and high pressure turbine, and finally, exiting the condenser into the TB. The licensee apportioned leakage from the MSL between the condenser and the high pressure turbine based on the bypass fraction of 0.005, but took no credit for the reduced flow from the condenser due to this apportioning. Consistent with containment leak rate reductions at 24 and 72 hours based on changing post-accident conditions, the licensee also reduced the leak rates for the MSIV release pathway to 60% at 24 hours and 50% at 72 hours. This reduction is allowed by the guidance of RG 1.183, and is therefore acceptable to the NRC staff.

For releases through this pathway, the licensee has taken credit for the mitigation of particulate radionuclide and elemental iodine activity. There are a number of mechanisms and processes used to model the mitigation and removal of the activity associated with these radionuclides, and in their LOCA analysis, the licensee considers the following:

- PC Spray Removal
- Natural Deposition in PC
- Impaction in MSL – Compact Streamline Impingement
- Impaction in MSL – Orifice Plugging
- Sedimentation in MSL and Condenser

The modeling of various mechanisms for radioactive particulate iodine removal, when more than one is used simultaneously for the same activity release in a dose analysis, should consider the effect of one model on the others. Although containment sprays, natural deposition, impaction, and sedimentation are all acting on the overall in-containment aerosol (and, indirectly, elemental for sprays and natural deposition) iodine source term, the total effect from each of these removal mechanisms is not the same as would be found by simply adding the removal coefficients for each model during a given time period. Therefore, as implemented for the design basis HNP LOCA analysis, the contiguous model used by the licensee to address this concern is found acceptable, as it accounts for series and parallel effects of each removal process. This is illustrated by the changing particulate geometric mean radius entering the MSLs and the changing aerosol mass mean diameter shown in the graphs provided to support the licensee's February 25, 2008 responses to RAIs 3 and 4, respectively.

The following subsections discuss the licensee's treatment of, and credit taken for, each of the activity removal mechanisms credited for the MSLs, as applicable to their design basis LOCA analysis, and the NRC staff evaluation of the licensee's modeling.

Impaction in MSLs

In the design basis LOCA analysis, the licensee reduced aerosol mass and activity, as well as elemental iodine activity, for releases through the MSLs and other lines with closed CIVs, by a decontamination factor (DF) of 2, attributed to a dynamic particulate phenomenon called impaction. To achieve this activity removal, the licensee credited the effects of a combination of two types of impaction taking place primarily at closed isolation valves, as described in their February 25, 2008 response to the NRC staff's RAI. The two types of impaction can be described as (1) compact streamline impingement and (2) orifice plugging. The licensee describes the first, compact streamline impingement, as removal of aerosol as it is entering a passage. The second, orifice plugging, can be described as plugging, or clogging, within and at the entrance to small passages due to an accumulation of removed particulate. Though the limiting HNP scenario that was analyzed assumed only one closed MSIV, it is of note, however, that if two closed MSIVs are modeled in one MSL, the effectiveness of impaction downstream of the MSIVs will be inherently reduced in that line, as the aerosol size distribution is also reduced. The licensee acknowledges that the compact streamline impingement type of impaction alone may not necessarily result in the credited DF of 2; however, the licensee contends that crediting the orifice plugging impaction alone can result in such removal. The basis for the licensee's contention is that, on a mass basis, only a small fraction of the leaked particulate would plug the postulated leak path, thereby closing the pathway opening and preventing any additional leakage from being released through that pathway. Considering this, the licensee maintains that a much larger DF, related to the complete clogging and prevention of any additional particulate from being transported through the MSIV, could be credited.

The NRC staff believes that, though there is merit to this plugging phenomenon and impaction in general, there is not enough empirical evidence, directly related to the unique and hypothetical conditions associated with a design basis LOCA event, to warrant full credit for such a considerable DF attributable to impaction. Therefore, the NRC staff does not find it acceptable to take full credit for impaction, as proposed by the licensee in its HNP LAR, when modeling removal of particulates in MSLs following a LOCA. However, the NRC staff does believe that the licensee has provided enough evidence to verify the conservatism of an associated DF of 2 in the specific design basis LOCA model at HNP. The contribution of this impaction DF to the overall iodine activity decontamination, does not lead to an excessive overall credit for iodine removal in the MSLs. When compared to an analysis of this pathway using a well-mixed model, such as that described and previously approved in AEB 98-03, the calculated activity removal in the MSLs and condenser would be analogous to that calculated here by the licensee. Therefore, the NRC staff found the overall iodine removal credited by the licensee to be acceptable, as modeled for HNP.

Sedimentation in MSLs and Condenser

The term sedimentation, or settling, refers to the gravity-driven phenomenon of particulate falling out of a gaseous suspension. The licensee credits sedimentation within the MSLs and condenser for the MSIV leakage contribution to PC leakage bypassing secondary containment. As mentioned above, the licensee uses a methodology implemented by a proprietary code to calculate aerosol activity removal as it is transported from the PC to the environment. This methodology accounts for the effect of upstream aerosol activity removal on the downstream capability of sedimentation removal in the MSLs and condenser. The NRC staff finds that the licensee's methodology results in a reasonable credit for aerosol activity removal in MSLs and condenser and is generally consistent with the calculation of such removal at various other plants using different models, including, but not limited to, the determination of activity removal efficiency based on the Monte Carlo assessment of aerosol settling velocities described in AEB 98-03. In addition, the presence of the HNP condenser in the modeled leakage pathway, with leakage being directed to the region below or within the tube sheet elevation, gives the NRC staff added confidence of activity removal and release holdup at least as substantial as that which was credited by the licensee. It was confirmed in the licensee's February 25, 2008 RAI response that all analyzed bypass pathways enter the condenser below the top of the tube sheet elevation. Therefore, because the overall removal credited by the licensee's model correlates well with AEB 98-03 and other previously approved models, the NRC staff finds their removal to be acceptable.

Other Bypass Leakage Pathways

The licensee states that, excluding MSIV leakage, 2.0% of the leakage from primary containment (L_a) is assumed to bypass secondary containment. The licensee has proposed a TS change to reflect this new leakage limit for HNP. The licensee modeled this leakage as a release directly to the condenser, then to the environment at a ground level through the plant vent. For the control room dose, the release from the condenser to the TB is transported to the control room by means of unfiltered inleakage. Other than sedimentation in the condenser, as was credited for MSIV leakage, the licensee conservatively credited no other activity removal mechanism for this pathway downstream of the drywell release. The licensee's model is conservative, and therefore, found to be acceptable to the NRC staff.

ESF Leakage

The licensee's model of ESF leakage conservatively assumed that, excluding noble gases, all isotopes that are released to the PC instantaneously transported to, and homogeneously mixed in, the torus water (suppression pool) at the onset of the gap activity release phase. This very conservative treatment is consistent with the guidance of RG 1.183. The HNP licensing basis does not have a prescribed limit on ESF leakage; however, the licensee assumed a considerably high leakage rate of 10 gpm (1.34 cfm). This leakage was assumed to begin coincident with the initiation of containment sprays and last for the duration of the accident. The licensee calculated the torus water temperature to be below 212 ºF; therefore, consistent with the guidance of RG 1.183, it was assumed that 10% of the iodine in the ESF leakage becomes airborne inside the RB, while all other elements remain in the water. Also consistent with the regulatory guidance, the iodine activity was assumed to be 97% elemental and 3% organic.

Direct Shine Dose

The following subsections discuss the licensee's evaluation of post-LOCA shine doses to control room personnel from the TB airborne activity cloud, ingress/egress through the TB to the control room, and other external shine sources.

TB Cloud Shine to the Control Room

As a result of this proposed amendment, the new licensing basis for HNP will assume that significant amounts of activity will be released into the TB following the postulated accident. In turn, this activity will result in dose consequences to control room operators by two mechanisms. The first, activity transport into the control room volume by unfiltered inleakage has been discussed in the previous sections; however, the second, shine dose from the airborne activity in the TB, is addressed here.

To calculate the shine dose consequence resulting from this source, the licensee first characterized the activity concentration in the TB airspace, using code output, resulting from the various activity transport mechanisms discussed in the earlier sections. The TB airspace volume and activity exhaust rate were conservatively limited to maximize the assumed concentration. The licensee credited the 2-foot control room wall for shielding. This model accurately and conservatively reflects the source-shield-receiver geometry applicable to HNP and is appropriate for the calculation of the shine dose from this TB cloud. Therefore, this treatment is acceptable to the NRC staff.

Ingress/Egress through the TB to the Control Room

The licensee assumed that the control room operator takes two trips through the TB each day for the 30-day accident duration. Using a walking speed of 3 miles per hour, and a travel distance of 321 feet, then adding a margin of 45 seconds, the licensee estimated the duration of each trip to be 2 minutes. The ingress/egress dose was then determined by multiplying the calculated time-dependent TB dose rate (determined from activity transport analysis) during each trip by the estimated exposure time of 2 minutes per trip. This methodology is reasonable based upon expected post-accident control room operator activities.

Other External Shine to the Control Room

As clarified in the February 25, 2008 RAI response, “other” external shine sources refers to the (1) TB external plume shine, (2) MSL and main condenser activity, (3) RB airborne cloud activity, (4) secondary containment door streaming considerations, and (5) main control room environmental control system filter.

TB external plume shine contribution was developed from the unshielded outside plume calculated at the TSC, which used the control room $\chi/_{Q}$ value. This value was then reduced assuming a factor of 10 per foot of concrete based on the dose reduction associated with 2-foot concrete control room walls and 6-inch TB walls. The NRC staff finds that this is an acceptable simplification based on the direct relationship between the shield thickness and resulting dose.

The MSL activity was modeled by an air-filled 2-foot diameter pipe with a 1.2-inch steel pipe wall, which was assumed to be shielded from the control room by 6.1 feet of concrete. MSL source terms were calculated as a function of time, using a computer code in the same manner as performed for the transport analysis. The condenser inventory as a function of time was calculated in the same way. The licensee then compared the condenser inventory to the MSL activities and determined it to be 4.22 times higher at 8 hours, 15.4 times higher at 24 hours, 44 times higher at 96 hours, and 264 times higher at 720 hours. The licensee took the dose from the condenser as the activity ratio times the MSL dose, and took no credit for the further distance from the control room or the additional slant path through the concrete shielding. This represents a conservative treatment by the licensee that does not challenge point-kernel code capability, and is therefore acceptable to the NRC staff.

The licensee modeled RB cloud shine as a rectangular slab shielded by 4 feet of concrete, which corresponds to a 2-foot control room wall and conservatively assumed 2-foot RB walls. The licensee ignored the added potential shielding effects of TB and RB internal structures. For this shine source, the licensee used source terms from an older core inventory, which is comprised of a different isotopic mix than that which was modeled for the AST analysis, but is still bounding. The NRC staff agrees that older TID-14844 source terms will bound the newly-calculated resulting AST source terms in the RB airspace, and finds this model acceptable.

The licensee modeled streaming through the RB door as a cylinder 36 feet long and having a 5.17-foot equivalent diameter, with the same activities as discussed above for the RB and a 2-foot concrete shield ignoring RB internal structures, including the 1-foot concrete door shield.

The licensee's model of this streaming source represents an acceptable and very conservative simplification that does not challenge point-kernel code capability.

For the control room environmental control system filter source, the licensee compared the assumed total accumulation of iodine (I-131 equivalent) activity on the filter to the code-calculated noble gas activity in the control room. This comparison showed that the accumulated iodine activity is well-bounded by the noble gas activity. Therefore, as noble gas serves as the primary contributor to whole body dose, the licensee used the calculated whole body dose to the control room as a surrogate for the unshielded dose expected from accumulated filter activity shine. The licensee then adjusted this dose to account for the 30-inch control room roof shielding. This treatment is conservative and, therefore, acceptable to the NRC staff.

The licensee disqualified the following potential control room shine dose contributors based on the available shielding between the source and control room occupants: TB ventilation filter, SGTS filter, and ECCS lines outside of the PC. The licensee states that a minimum of 4.5 feet of concrete is designed between the aforementioned filters, and that at least 6 feet of concrete exists between the ECCS lines outside of the PC. This degree of shielding is sufficient to attenuate the amount of fission product gamma activity expected to accumulate on filters and in the core coolant following the postulated design basis LOCA. Therefore, the NRC staff agrees with the licensee's assertion that these potential contributors will be negligible based upon the available concrete shielding.

Technical Support Center (TSC) Dose Consequence Assessment

In the LAR submittal, the licensee clearly shows that the TSC 30-day inhalation and immersion dose was thoroughly analyzed for the LOCA and each of the other DBAs. In examining their post-accident TSC dose consequences, the licensee finds that the 30-day doses do not exceed 5 rem TEDE. The licensee's analyses indicate that they comply with the regulatory requirements for the TSC as given in NUREG-0737 and Paragraph IV.E.8 of Appendix E to 10 CFR Part 50. Therefore, the licensee's examination of the DBA dose consequences to the TSC is acceptable to the NRC staff.

Summary

The licensee concluded that the radiological consequences at the EAB, LPZ, and control room are within the dose criteria specified in 10 CFR 50.67 and the accident-specific dose acceptance criteria specified in SRP 15.0.1 and RG 1.183. These accident-specific dose acceptance criteria for the LOCA are a TEDE of 25 rem at the EAB for any two hours, 25 rem at the outer boundary of the LPZ for the duration of the accident, and 5 rem for access to, and occupancy of, the control room for the duration of the accident. The NRC staff finds that the licensee used conservative analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.1.1 of this SE and with those stated in the HNP FSAR as design bases. The NRC staff also performed independent calculations of the dose consequences of the postulated LOCA releases, using the licensee's assumptions for input to the RADTRAD computer code. The NRC staff's calculations confirmed the licensee's dose results. The major parameters and assumptions used by the licensee, and found acceptable to the NRC staff, are presented in Table 3.2.1. The results of the licensee's design basis radiological consequence calculation are provided in Table 3.2. The EAB, LPZ, and control room doses estimated by the licensee for the LOCA were found to meet the applicable accident dose acceptance criteria and are therefore acceptable.

2.1.2.3 Fuel Handling Accident

The current HNP design basis FHA analysis is based on the traditional accident source term described in TID-14844, "Calculation of Distance Factor for Power and Test Reactor Sites." The HNP licensing basis analysis is presented in FSAR Section 15.3.5, "Fuel Handling Accident." To support implementation of the AST, as requested by the subject LAR, the licensee reanalyzed the offsite and control room radiological consequences of the postulated FHA. This reanalysis was performed to demonstrate that the ESFs designed to mitigate the radiological consequences at HNP will remain adequate after implementation of the AST.

The licensee described the AST-based reanalysis of the FHA in the safety analysis submitted as part of the LAR. Included in this assessment are the key assumptions, parameters, and newly calculated offsite and control room doses associated with implementing the AST methodology. The licensee cites RG 1.183 as providing the primary radiological analysis assumptions for their reanalysis of the postulated design basis FHA. Specifically, the NRC staff's guidance for analysis of the FHA is detailed in Appendix B of RG 1.183.

As analyzed for HNP, the postulated FHA involves a drop of a fuel assembly on top of other fuel assemblies in the reactor core during refueling operations. The licensee has determined that the drop distance associated with this location bounds the maximum height that is allowed by the HNP refueling equipment configuration and this is the limiting case because it results in the maximum release of fission products to the reactor building. Also, the licensee has determined damage due to a fuel assembly drop over the core into the reactor vessel bounds a drop in the spent fuel pool. All fuel types currently stored in the spent fuel pool are bounded by this analysis.

Activity Source

For the FHA analysis, the licensee assumed that the core isotopic inventory available for release into the containment is based on maximum full power operation of the core at 2,818 megawatts thermal (MWt), or 1.005 times the current licensed thermal power level of 2,804 MWt, in order to account for the ECCS flow uncertainty. In addition, the licensee increased the core isotopic activities by 10% to allow for possible future power uprates. The burnup and enrichment parameters assumed when determining the core isotopic inventory are within current licensed limits for fuel at HNP.

The design basis radial peaking factor of 1.50, as shown in the HNP FSAR, was applied to the isotopic activity for the damaged fuel assemblies. The fraction of core isotopic activity assumed to be available for release from the gap of failed fuel (i.e., fuel experiencing cladding failure as a result of the drop) is provided in Table 3 of RG 1.183, and was used in the licensee's DBA analysis. These gap fractions are 5% for noble gases and iodines, except for Kr-85 and I-131, where 10% and 8% are assumed, respectively, and 12% for alkali metals. Consistent with the guidance of RG 1.183, all iodine activity released from the coolant in the reactor vessel was assumed to be of elemental and organic chemical form, in the ratio of 99.85% and 0.15%, respectively. No particulate forms of activity are assumed to be released. All particulate is assumed to be retained by the water in the pool or reactor cavity (i.e., an infinite DF). Also, the licensee states that HNP has no fuel that exceeds the burnup limit assumption expressed in Footnote 11 of RG 1.183. Further, in their RAI response of February 25, 2008, the licensee states that, "Appropriate measures will be implemented to assure that HNP will operate in compliance with the fuel burnup parameters delineated in Footnote 11 to Table 3 of RG 1.183." Therefore, the total assumed activity in the fuel gap, and available for release following the postulated FHA drop is found to be acceptable to the NRC staff.

As a design basis, the licensee assumed 24 hours of decay for the accident analysis, corresponding to the time before any movement of fuel can be initiated; the movement of fuel, or fuel handling, before this period would be unanalyzed and not consistent with the assumed design basis. Therefore, activity available for release was calculated to correspond to this post-shutdown decay time. The licensee assumed that 172 fuel rods will be damaged as a result of the postulated FHA, and thus instantaneously release all of their available gap activity to the environment, taking no credit for RB closure or isolation. HNP currently uses only GE14 10x10 fuel, and has performed the FHA dose consequence analysis assuming a drop and impact of this fuel type. Therefore, as a design basis, the characteristic fuel damage and activity release resulting from the postulated FHA is associated with only this GE14 fuel type.

Transport Methodology and Assumptions

As analyzed for HNP, the postulated FHA involves a drop of a fuel assembly on top of other fuel assemblies in the core during refueling operations. Even though the most limiting drop height and subsequent fuel damage is associated with a drop over the core, for conservatism, the licensee assumed the water coverage and DF associated with the spent fuel pool. The minimum water coverage allowed by HNP TS 3.7.8 is 21 feet, which is less than the 23-foot water coverage required to assume an overall DF of 200 in accordance with RG 1.183, Appendix B. The licensee calculated an adjusted DF of 142 associated with the 21-foot water depth, based upon assuming an overall iodine DF of 286 (determined from the guidance of RG 1.183). The licensee assumed that, because RG 1.183 allows for an overall DF of 200 for 23 feet of water coverage, this DF of 286 represents a factor of conservatism of 1.43, or 286/200. When this factor is applied to the DF calculated assuming 21 feet of water coverage, the licensee determined an adjusted overall iodine DF of 142. This credited iodine DF is acceptable, because the methods used by the NRC staff calculate an iodine DF that very closely compares to the licensee's resulting DF.

Noble gas activity is assumed to be released from the reactor vessel water without experiencing any reduction. The DF calculation by the licensee is conservative and acceptable to the NRC staff, as it is consistent with the guidance of RG 1.183.

The licensee analyzed two cases for the analysis of post-FHA activity transport:

- Case 1: For this case, the licensee assumed the 120-second drawdown time. Prior to that time, the licensee assumed that airborne activity is released, unfiltered, and at ground level. After 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% efficiency for particulates and all forms of iodine.

- Case 2: For this case, 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.

Both activity transport cases are modeled conservatively and the dose consequences for both cases have been determined to meet applicable acceptance criteria. Therefore, the licensee's model is acceptable to the NRC staff.

Summary

The licensee concluded that the radiological consequences at the EAB, LPZ, and control room are within the dose criteria specified in 10 CFR 50.67 and the accident-specific dose acceptance criteria specified in SRP 15.0.1 and RG 1.183. These accident-specific dose acceptance criteria for the FHA are a TEDE of 6.3 rem at the EAB for any two hours, 6.3 rem at the outer boundary of the LPZ, and 5 rem for access to, and occupancy of, the control room for the duration of the accident. The NRC staff finds that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.1.1 of this SE and with those stated in the HNP FSAR as design bases. The major parameters and assumptions used by the licensee, and found acceptable to the NRC staff, are presented in Table 3.2.2. The results of the licensee's design basis radiological consequence calculation are provided in Table 3.2. The EAB, LPZ, and control room doses estimated by the licensee for the FHA accident were found to meet the applicable accident dose acceptance criteria and are therefore acceptable.

2.1.2.4 <u>Main Steam Line Break (MSLB) Accident</u>

The current HNP design basis MSLB accident analysis is based on the traditional accident source term described in TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites." The HNP licensing basis analysis is presented in FSAR Section 15.3.4, "Steam System Piping Break Outside Containment." To support implementation of the AST, as requested by the subject LAR, the licensee reanalyzed the offsite and control room radiological consequences of the postulated MSLB. This reanalysis was performed to demonstrate that the ESFs designed to mitigate the radiological consequences at HNP will remain adequate after implementation of the AST.

The licensee described the AST-based reanalysis of the MSLB accident in the safety analysis submitted as part of the LAR. Included in this assessment are the key assumptions, parameters, and newly calculated offsite and control room doses associated with implementing the AST methodology. The licensee cites RG 1.183 as providing the primary radiological analysis assumptions for their reanalysis of the postulated design basis MSLB accident. Specifically, the NRC staff guidance for analyses of the MSLB accident is detailed in Appendix D of RG 1.183.

The design basis MSLB accident is generally defined as an instantaneous circumferential break of one main steam line outside the secondary containment, downstream of the outside isolation valve. It is assumed that pipe end displacement due to this double ended guillotine break is such that the maximum blowdown rate is permitted to occur. The licensee assumed that the break flow is terminated by closure of the MSIVs, and that the coolant mass released through the break includes the line inventory plus the system mass released through the break prior to isolation. The radiological consequences of a main steam line break outside secondary containment will bound the consequences of a break inside containment. Thus, it is acceptable to only consider an MSLB outside of containment with regard to radiological consequences. In addition, this accident is postulated to occur non-mechanistically and without an identified cause in order to evaluate consequences of a hypothetical large steam line rupture.

Activity Source

For the design basis MSLB, the licensee assumed no fuel failure, consistent with the current HNP licensing basis, because the temperature and pressure transients resulting from this event are not severe enough cause such failures at HNP. To determine the maximum offsite and control room dose, the licensee assumed that a reactor transient, or iodine spike, has occurred prior to the postulated MSLB and has raised the coolant iodine concentration to 10 times the HNP TS maximum allowable coolant equilibrium iodine concentration for continued operation. This is done as a design basis, and in accordance with the guidance of RG 1.183, when no fuel failure is assumed.

The HNP TS 3.4.6 maximum allowable coolant equilibrium iodine activity concentration for continued operation assumed in the MSLB analysis is 0.2 µCi/gm Dose Equivalent (DE) I-131. The postulated pre-accident iodine spike raises this value to 2.0 µCi/gm DE I-131, which is consistent with the TS change proposed by the subject LAR that specifies that the maximum coolant iodine concentration be limited to this value. Though the licensee analyzed both the pre-accident iodine spike and equilibrium concentration cases, the spiked activity case bounds, because, as shown in Table 3.2, the offsite dose consequence resulting from this activity meets the lower acceptance criterion for the equilibrium activity case that is suggested in Table 6 of RG 1.183. Also, because the radiological consequences are directly related to the coolant activity released, and since the equilibrium concentration case has a lower coolant activity release than the iodine spike case, the equilibrium concentration case would always result in lower offsite dose consequences for the licensee's specific and acceptable DBA dose model. For control room doses, however, the licensee states that the pre-accident iodine spike case bounds the equilibrium concentration case, but does not report the resulting doses from the equilibrium case. Therefore, the NRC staff dos not endorse removing an assessment of control room dose consequences resulting from the equilibrium concentration case from HNP licensing basis.

The licensee also included noble gas activity in their total MLSB accident release. The HNP TS delayed offgas release rate limit is 0.240 Ci/sec. However, as shown in their LAR, the licensee assumed a 0.3 Ci/sec release rate that will bound the TS value. The licensee took no credit for decay by assuming that the core noble gas activity release rates are the same as from the break.

Consistent with RG 1.183 guidance, the licensee assumed that the speciation of radioactive iodine released from failed fuel is 95% aerosol (particulate), 4.85% elemental, and 0.15% organic; whereas, the speciation of radioactive iodine released by coolant blowdown is 97% elemental and 3% organic. Because no fuel failure was assumed, the coolant iodine speciation was used for this DBA analysis. The licensee also considered the maximum TS noble gas and cesium activity to be available for release from the steam blowdown and coolant, respectively. The licensee determined the cesium contribution to be negligible to the total dose consequence of the design basis MSLB accident. The licensee's treatment of cesium is therefore conservative, with respect to the RG 1.183 guidance, and acceptable to the NRC staff, because the applicable guidance does not explicitly suggest that cesium activity be considered as a dose contributor.

Transport Methodology and Assumptions

The design basis MSLB accident is generally defined as an instantaneous circumferential break of one main steam line outside the secondary containment, downstream of the outside isolation valve. It is assumed that pipe end displacement due to this double ended guillotine break is such that the maximum blowdown rate is permitted to occur. The licensee assumed that the break flow is terminated by closure of the MSIVs, and that the coolant mass released through the break includes the line inventory plus the system mass released through the break prior to isolation. The radiological consequences of a main steam line break outside secondary containment will bound the consequences of a break inside containment. Thus, it is acceptable to only consider an MSLB outside of containment with regard to the radiological consequences.

Consistent with the current HNP licensing basis, the licensee assumed break isolation in 5.5 seconds, corresponding to the maximum MSIV closing time of 5 seconds, plus an assumed closure signal delay time of 0.5 seconds. The licensee took no credit for reduction in break flow as the valves are closing. In Table 30 of Enclosure 1 to the August 29, 2006 LAR, the licensee presented the coolant blowdown rates as a function of time until isolation. However, this table was supplemented by the discussion in the February 25, 2008 RAI response (ADAMS Accession No. ML080570185). The discussion in the RAI response explains that the actual total mass release was calculated by the integration over the curve shown therein. This integration shows that the total coolant mass release was calculated to be 5.25 E+04 lbm, consisting of 1.35 E+04 lbm of steam and 3.91 E+04 lbm of a steam liquid mixture at 7% quality (HNP design basis assumption). When applying the assumed steam quality and calculated liquid flashing fraction (42% assuming a constant enthalpy process) to the mixture, the total mass of airborne activity-carrying steam released is equal to 3.16 E+04 lbm.

For assessment of the offsite dose the licensee assumed that, following accident initiation, the radionuclide inventory from the released coolant reaches the environment instantaneously, taking no credit for holdup in the turbine building. The release modeled by the licensee was assumed to be in the form of an instantaneous "puff" of steam activity that results from the released mass of coolant. The appropriate ground level release EAB and LPZ χ/Q values were applied to this release for the 30-day accident duration. However, consistent with regulatory guidance, EAB doses were only assessed for the first 2 hours. The NRC staff finds the use of this puff release model to be acceptable because of the very short duration of the MSLB release and inherent conservatism of the instantaneous release assumed by the licensee.

The HNP control room is located within the turbine building; therefore, it is conservative to assume that the activity carried by the MSLB steam release is held up in the TB and drawn into the control room by the assumed unfiltered inleakage rate, as opposed to applying a χ/Q value associated with a release to the environment. The licensee modeled their post-accident activity transport to the control room in this manner. Ingress and egress doses were also calculated for control room operators who pass through the TB to enter and exit the control room in the same manner as was discussed above in this SE.

Summary

The licensee concluded that the radiological consequences at the EAB, LPZ, and control room are within the dose criteria specified in 10 CFR 50.67 and the accident-specific dose acceptance criteria specified in SRP 15.0.1 and RG 1.183. These accident-specific dose acceptance criteria for the MSLB accident, assuming a pre-accident iodine activity spike and no fuel failure, are a TEDE of 25 rem at the EAB for any two hours, 25 rem at the outer boundary of the LPZ, and 5 rem for access to, and occupancy of, the control room for the duration of the accident. For an MSLB assuming the equilibrium iodine concentration, the acceptance criteria are a TEDE of 2.5 rem at the EAB for any two hours, 2.5 rem at the outer boundary of the LPZ, and 5 rem for access to, and occupancy of, the control room for the duration of the accident. The NRC staff finds that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.1.1 of this SE and with those stated in the HNP FSAR as design bases. The major parameters and assumptions used by the licensee, and found acceptable to the NRC staff, are presented in Table 3.2.3. The results of the licensee's design basis radiological consequence calculation are provided in Table 3.2. The EAB, LPZ, and CR doses estimated by the licensee for the MSLB accident were found to meet the applicable accident dose acceptance criteria and are therefore acceptable.

2.1.2.5 Control Rod Drop Accident

The current HNP design basis CRDA analysis is based on the traditional accident source term described in TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites." The HNP licensing basis analysis is presented in FSAR Section 15.3.2, "Control Rod Drop Accident." To support implementation of the AST, as requested by the subject LAR, the licensee reanalyzed the offsite and control room radiological consequences of the postulated CRDA. This reanalysis was performed to demonstrate that the ESFs designed to mitigate the radiological consequences at HNP will remain adequate after implementation of the AST.

The licensee described the AST-based reanalysis of the CRDA in the safety analysis submitted as part of the LAR. Included in this assessment are the key assumptions, parameters, and newly calculated offsite and control room doses associated with implementing the AST methodology. The licensee cites RG 1.183 as providing the primary radiological analysis assumptions for their reanalysis of the postulated design basis CRDA. Specifically, the NRC staff guidance for analysis of the CRDA is detailed in Appendix C of RG 1.183.

Activity Source

For the CRDA analysis, the licensee assumed that the core isotopic inventory available for release into the containment, is based on maximum full power operation of the core at 2,818 megawatts thermal (MWth), or 1.005 times the current licensed thermal power level of 2,804 MWth, in order to account for the ECCS flow uncertainty. In addition, the licensee increased the core isotopic activities by 10% to allow for possible future power uprates. The burnup and enrichment parameters assumed when determining the core isotopic inventory are within current licensed limits for fuel at HNP.

The maximum core radial peaking factor of 1.50 was applied to the isotopic activity for damaged rods. The fraction of core isotopic activity assumed to be available for release from the gap of failed fuel (i.e., fuel with cladding perforation) is provided in RG 1.183, Table 3 and Appendix C, Section 1. These gap fractions were used in the licensee's DBA analysis. These gap fractions are 10% for noble gases and iodines, 12% for alkali metals, and 5% for other halogen isotopes. Consistent with RG 1.183 guidance, the licensee assumed that the speciation of radioactive iodine released from failed fuel is 95% aerosol (particulate), 4.85% elemental, and 0.15% organic, and that all other non-noble gas isotopes are released in 100% particulate form. Also, the licensee assumed that there is no fuel exceeding the burnup limit assumption expressed in Footnote 11 of RG 1.183. Therefore, the total assumed activity in the fuel gap, and available for release following the postulated CRDA is found to be acceptable to the NRC staff.

The failed fuel activity release for the design basis CRDA was characterized by the licensee's calculation that 1189 fuel rods experience cladding failure, and that 11 rods (0.0225 % of total core) experience melt, following the postulated CRDA. Therefore, this assumption based on the currently used GE14 10x10 fuel type, is the new HNP design basis assumption for the CRDA, and will bound the expected damage to all other fuel types currently in use at the plant. The licensee states that, as noted in HNP FSAR Section 15.3.2.4, when operating within the constraints of the BPWS, a CRDA will not exceed the 280 calories/gram (cal/g) design criterion. BPWS plants have been statistically analyzed and show that, in all cases, the peak fuel enthalpy in a CRDA would be much less than the 280 cal/g design limit, and thus, the CRDA has been deleted from the standard GE BWR reload package for BPWS plants. Therefore, the NRC staff agrees that the assumption of 1189 failed fuel rods and 11 rods experiencing melt is conservative.

Transport Methodology and Assumptions

The licensee has defined the design basis CRDA as the rapid removal of the highest worth control blade from the core due to a decoupling of the control rod drive mechanism from a cruciform control blade. In turn, this results in a reactivity excursion that encompasses the consequences of other postulated CRDAs. In their FSAR, the licensee states that HNP is a banked position withdrawal sequence (BPWS) plant. The GESTAR generic CRDA analysis demonstrates that the accident does not result in fuel melting for BPWS plants. However, for the purpose of this analysis, the licensee did assume some fuel damage (i.e., cladding perforation and fuel melting) to occur. The HNP AST analysis for the CRDA considers two scenarios with regard to the activity release and transport pathways, as follows:

1. Case 1: For this case, the licensee assumed that activity is released from the turbine/condenser and is held up in the TB so that control room doses are calculated conservatively, since the control room is located in the TB. The TB then exhausts to the environment starting at 9 hours with a flow of 15,000 cfm via the RB vent. Ingress and egress doses were also calculated for control room operators who pass through the TB to enter and exit the control room in the same manner as was discussed above in this SE.

2. Case 2: For this case, the licensee assumed that activity is released from the turbine/condenser and is not held up in the TB; instead it is released directly to the environment at ground level to conservatively maximize offsite doses (i.e., doses to the EAB and LPZ). In addition, for offsite doses, the licensee evaluated the mechanical vacuum pump (MVP) forced flow path from the turbine/condenser. The MVP normally discharges to the plant stack through the gland-seal holdup line, which provides holdup for up to 2 minutes, but provides no filtration. The pump trips on high MSL radiation, but

the licensee conservatively assumed the release to continue for 24 hours. The MVP flow was assumed to be 2,200 cfm for 24 hours, at which time the release was assumed to terminate, consistent with the guidance of RG 1.183.

Consistent with the guidance of RG 1.183, the licensee assumed that 100% of the noble gas, 10% of the iodine, and 1% of the remaining radionuclides reach the turbine and condensers, and of that activity, 100% of the noble gas, 10% of the iodine, and 1% of the particulate radionuclides are available for release to the environment.

In the February 25, 2008 submittal regarding the potential for post-CRDA releases from other forced flow paths, such as the steam jet air ejector (SJAE), the licensee states that, in response to a CRDA and subsequent reactor scram, operation of the SJAEs is terminated by terminating steam to the SJAEs. Therefore, the SJAEs would not contribute to a post-accident dose. The licensee also states that the doses resulting from the MVP release would bound that of doses from an SJAE release due to the lower release rate, more favorable dispersion, and filtration associated with the SJAE pathway. The NRC staff agrees with this assessment, and finds the exclusion of this potential leakage path acceptable for the design basis CRDA analysis at HNP.

Summary

The licensee concluded that the radiological consequences at the EAB, LPZ, and control room are within the dose criteria specified in 10 CFR 50.67 and the accident-specific dose acceptance criteria specified in SRP 15.0.1 and RG 1.183. The accident-specific dose acceptance criteria for the CRDA at HNP are a TEDE of 6.3 rem at the EAB for any two hours, 6.3 rem at the outer boundary of the LPZ, and 5 rem for access to, and occupancy of, the control room for the duration of the accident. The NRC staff finds that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.1.1 of this SE and with those stated in the HNP FSAR as design bases. The NRC staff also performed an independent calculation of the dose consequences of the licensee's CRDA using the licensee's assumptions for input to the RADTRAD computer code. The NRC staff's calculation confirmed the licensee's dose results. The major parameters and assumptions used by the licensee, and found acceptable to the NRC staff, are presented in Table 3.2.4. The results of the licensee's design basis radiological consequence calculation are provided in Table 3.2. The EAB, LPZ, and CR doses estimated by the licensee for the CRDA were found to meet the applicable accident dose acceptance criteria and are therefore acceptable.

2.1.2.6 Control Room Habitability and Modeling

The current HNP DBA analyses, as described in FSAR Chapter 15, do not calculate control room dose, therefore, the control room dose model provided in the revised DBA accident analyses that support this AST-based LAR, represents a change in the HNP licensing basis.

For their revised analyses where control room isolation and/or filtration is credited, the licensee assumed a Main Control Room Environmental Control (MCREC) system intake flow rate of 250 cfm, 2100 cfm of recirculation flow, and 95% filtration efficiency for elemental iodine, organic iodine, and all particulate forms of radionuclide activity. The licensee also assumed that there was no delay in the initiation of the MCREC system, and that the associated filtration was available from the onset of activity release for all accidents, consistent with the current design basis.

Unfiltered Inleakage

The current HNP licensing basis control room unfiltered inleakage limit is 110 cfm based on the administration of KI tablets to control room occupants within 2 hours after the start of the design basis LOCA. The HNP Units 1 and 2 common control room, as part of the control building, is located between the open end bays of the HNP Units 1 and 2 TBs. The majority of the ductwork associated with the MCREC system, which encompasses two independent filter trains for post-accident pressurization of the control room, is located external to the control room boundary, on top of the control building, and within the confines of the HNP Units 1 and 2 TBs.

By letters dated August 4, 2003, March 29, 2004, October 27, 2004, and November 10, 2005, SNC submitted a course of action for developing responses to NRC Generic Letter (GL) 2003-01, “Control Room Habitability” information requests for HNP. GL 2003-01 was written to inform licensees that the design basis assumptions used for control room unfiltered inleakage, even with a pressurized control room, could be non-conservative. This was validated through testing at several power reactor facilities using the standard test method described in American Society for Testing and Materials (ASTM) consensus standard E741, “Standard Test Method for Determining Air Change in a Single Zone by Means of a Tracer Gas Dilution.”

In order to address the possibility of unfiltered inleakage into the HNP control room, the incorporation of KI was approved on an interim basis as a measure to limit the thyroid dose to control room occupants in the event of a design basis LOCA. The incorporation of KI in the interim licensing basis was provided to assure that the 30-day thyroid dose remains within the regulatory limits of 10 CFR 50, Appendix A, GDC 19, with control room unfiltered inleakage of up to 110 cfm. As a condition of the licensing basis, the crediting of KI in limiting post-LOCA dose consequences to control room personnel was for an interim period, expiring on May 31, 2010.

According to the licensee, tracer gas testing of the HNP control room envelope was completed in June 2006 using ASTM consensus standard E741. The licensee states that the most limiting results from testing revealed 5 cfm unfiltered inleakage into the control room. The licensee indicates that, with the completion of tracer gas testing, they will be completing their response to GL 2003-01 under a separate letter.

The licensee deemed it necessary to change to the HNP licensing basis by implementing the AST in order to comply with control room habitability regulatory requirements without relying on the KI interim licensing basis. Therefore, approval of this proposed AST LAR will ensure that the design basis radiological analyses for occupants of the control room reflects the most limiting unfiltered inleakage into the control room. For all DBA dose consequence analyses, the licensee has assumed an unfiltered inleakage of 115 cfm into the control room (which is significantly larger than the recently calculated 5 cfm), except in the case of the FHA, where a limiting 10,000 cfm of unfiltered inleakage is assumed. According to the licensee’s measurements, there is significant margin between the measured unfiltered inleakage and the unfiltered inleakage assumptions used in the LOCA dose consequence analysis, which is the most limiting DBA at HNP for occupants of the control room.

2.1.3 Technical Specification Changes

Revision to the TS 1.1 Definition of "*Dose Equivalent I-131*"

The licensee has proposed to revise the definition of *Dose Equivalent I-131* to HNP Units 1 and 2 TS Section 1.1. The licensee's revised DBA dose consequence analyses use committed effective dose equivalent dose conversion factors from Federal Guidance Report (FGR) 11, ORNL, 1988, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," as the source of dose conversion factors instead of the current TID-14844, RG 1.109, Rev. 1, and ICRP 30 referenced dose conversion factors.

With the implementation of AST, the previous whole body and thyroid dose guidelines of 10 CFR 100.11 and 10 CFR 50, Appendix A, GDC 19, are replaced by the total effective dose equivalent (TEDE) criteria of 10 CFR 50.67(b)(2). This new definition reflects adoption of the dose conversion factors and dose consequences of the revised radiological analyses. Thus, this proposed revision to the definition of *Dose Equivalent I-131* is supported by the justification for the proposed licensing basis revision to implement the AST, and conforms to the implementation of the AST and the TEDE criteria in 10 CFR 50.67. Therefore, the NRC staff finds the proposed revision to the TS 1.1 definition *Dose Equivalent I-131* acceptable.

Revision to TS Section 3.4.6, "Reactor Coolant System Specific Activity"

The licensee has proposed to revise the reactor coolant radioactivity concentration limit specified in HNP Units 1 and 2 TS Section 3.4.6 from 4.0 μCi/gm DE I-131 to 2.0 μCi/gm DE I-131. This revision is consistent with the reactor coolant activity concentration assumed in the design basis MSLB accident analysis as described in Section 2.1.2.4 above. The licensee's analysis of the MSLB accident used a source term based on the maximum short-term reactor coolant specific activity of 2.0 μCi/gm DE I-131 and resulted in calculated radiological consequences, shown in Table 3.2, that are below the applicable acceptance criteria, as discussed in Section 2.1.1.

From the TS requirement and safety perspective, this proposed revision will implement a limit that is more conservative than the existing requirement. Therefore, the NRC staff finds the proposed revision to the HNP Units 1 and 2 TS 3.4.6 limit to be acceptable with respect to the radiological consequences of DBA.

Revision to TS Section 3.6.1.3, Surveillance Requirements (SR), Secondary Containment Bypass Leakage

For HNP Unit 1, the licensee has proposed to add an SR limiting the total allowed secondary containment bypass leakage, for all secondary containment bypass paths, to a maximum of 2.0% of the maximum PC leakage rate. For HNP Unit 2, a similar SR currently exists, so the licensee has proposed to revise the current value of 0.9% to the new value of 2.0%, which will be consistent with the HNP Unit 1 change.

The licensee's analysis of the design basis LOCA assumed this maximum secondary containment bypass leakage in calculation of control room and offsite radiological dose consequences. The resulting doses, as shown in Table 3.2, are below the applicable acceptance criteria, as discussed in Section 2.1.1. Therefore, the licensee's new analyses of record show that, as a design basis, these proposed HNP TS revisions are acceptable with respect to the radiological consequences of DBAs.

Revision to TS Section 3.6.1.3, SR, MSIV Leakage

The licensee has proposed to increase the HNP Unit 1, and decrease the HNP Unit 2, maximum allowable combined MSIV leakage rate. Currently, for Unit 1 the TS SR specifies that the leakage through each MSIV be ≤ 11.5 scfh when tested at ≥ 28.0 psig, and for Unit 2 the TS SR specifies that the leakage be ≤ 100 scfh per line and 250 scfh combined total when tested at ≥ 28.8 psig. The proposed revision will change both of these SRs to have a maximum combined total of 100 scfh, with no limit per line.

The licensee's analysis of the design basis LOCA assumed this maximum MSIV leakage in calculation of control room and offsite radiological dose consequences. The resulting doses, as shown in Table 3.2, are below the applicable acceptance criteria, as discussed in Section 2.1.1. Therefore, the licensee's new analyses of record show that, as a design basis, these proposed HNP TS revisions are acceptable with respect to the radiological consequences of DBAs.

Revision to TS Section 3.6.2.5, RHR Drywell Spray

The licensee has proposed to incorporate a TS to reflect the credit taken for drywell sprays in the model of activity release mitigation for the analysis of the design basis LOCA, as discussed. As it is confirmed that the licensee did indeed credit drywell sprays for the HNP design basis LOCA analysis, it is necessary for the approval of this proposed amendment that a TS governing the operability and surveillance requirements associated with these sprays be included in both the HNP Unit 1 and Unit 2 TSs.

Summary

As described above, the NRC staff reviewed the assumptions, inputs, and methods used by the licensee to assess the radiological consequences of the postulated DBA analyses with the proposed TS changes. The NRC staff finds that the licensee used analysis methods and assumptions consistent with the conservative regulatory requirements and guidance and the NRC staff compared the doses estimated by the licensee to the applicable criteria as discussed above. The NRC staff also finds, with reasonable assurance, that the licensee's estimates of the Control Room, EAB, and LPZ doses will comply with these criteria. The NRC staff further finds, with reasonable assurance, that HNP, 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 the DBAs.

Table 3.1.1

HNP Control Room Atmospheric Dispersion Factors ($^{\chi}/_{Q}$, sec/m^3)

Time Interval	Reactor Building Vent	Main Stack
0-2 hours	1.41×10^{-3}	3.76×10^{-6}
2-8 hours	1.08×10^{-3}	2.88×10^{-6}
8-24 hours	4.70×10^{-4}	7.50×10^{-7}
1-4 days	3.54×10^{-4}	7.67×10^{-7}
4-30 days	2.67×10^{-4}	5.04×10^{-7}

Table 3.1.2

HNP EAB and LPZ Atmospheric Dispersion Factors ($^{\chi}/_{Q}$, sec/m^3)

Time Interval	Ground	Elevated
EAB (1250 m)		
0-2 hours	3.1×10^{-4}	1.7×10^{-6}
LPZ (1250 m)		
0-2 hours	3.1×10^{-4}	1.7×10^{-6}
2-8 hours	1.7×10^{-4}	9.4×10^{-7}
8-24 hours	2.3×10^{-5}	3.9×10^{-7}
1-4 days	1.1×10^{-5}	2.0×10^{-7}
4-30 days	4.5×10^{-6}	8.0×10^{-8}

Table 3.2

Licensee Calculated Radiological Consequences of Design Basis Accidents at HNP

Design Basis Accident	Control Room		EAB[a]		LPZ	
	Total Dose[b] (rem TEDE)	Acceptance Criteria (rem TEDE)	Total Dose[c] (rem TEDE)	Acceptance Criteria (rem TEDE)	Total Dose[d] (rem TEDE)	Acceptance Criteria (rem TEDE)
LOCA	4.90E+00	*5.0*	3.40E-01	*25*	7.50E-01	*25*
FHA						
Case 1	7.20E-01	*5.0*	2.50E-01	*6.3*	2.50E-01	*6.3*
Case 2	3.50E+00	*5.0*	1.20E+00	*6.3*	1.20E+00	*6.3*
MSLB						
Case 1	3.90E+00	*5.0*	1.50E-01	*25*	1.50E-01	*25*
Case 2	N/A	*5.0*	1.50E-02	*2.5*	1.50E-02	*2.5*
CRDA	3.80E+00	*5.0*	3.33E-01	*6.3*	5.40E-01	*6.3*

[a] The licensee calculated the EAB dose for the worst 2-hour period of the accident.
[b] The licensee's control room dose results have been rounded to three significant digit precision.
[c] The licensee's EAB dose results have been rounded to three significant digit precision.
[d] The licensee's LPZ dose results have been rounded to three significant digit precision.

Table 3.2.1

Key Parameters Used in Radiological Consequence Analysis of Loss of Coolant Accident

Parameter	Value
Reactor Core Power, MWth	2818
Primary Containment Volume, ft^3 Drywell Airspace Torus Airspace (minimum) Suppression Pool (minimum)	 146,010 109,900 85,110
Secondary Containment/RB Volume, ft^3	1,300,000
Spray Delay time, min	15
Drywell to Torus Airspace Mixing Profile, cfm 0.00 – 2.03 hours 2.03 – 2.06 hours 2.06 – 2.39 hours 2.39 – 3.00 hours 3.00 – 720 hours	 0 26,457 685 349 0
Primary Containment Leakage Rate, weight % per day 0 – 24 hours 24 – 72 hours 72 hours – 30 days	 1.20 0.72 0.60
Secondary Containment Drawdown Time, min	2
RB Volume Credited for Dilution (50% of total), ft^3	650,000
RB SGTS Exhaust Rate, cfm	8000
RB SGTS Filtration Efficiency, % Noble Gas All other radionuclides	 0 95
MSL Volume, ft^3	392

Table 3.2.1 cont'd

MSIV Leakage Rate (Initial) Drywell to MSL Standard, scfh True Volumetric, cfh MSL to Condenser Standard, scfh True Volumetric, cfh	 100 263 100 49.7
Low Pressure Turbine/Condenser Volume, ft^3	172,000
TB Free Volume, ft^3	6,500,000
TB Fan Initiation Time (manual), hrs	9
TB Fan Exhaust Rate, cfm	15,000
ESF Leakage Rate, gpm	10
ESF Leakage Iodine Re-Evolution, %	10
ESF Leakage Iodine Release Speciation, % Elemental Organic	 97 3
Atmospheric Dispersion Factors	Tables 3.1.1 and 3.1.2

Table 3.2.2

Key Parameters Used in Radiological Consequence Analysis of Fuel Handling Accident

Parameter	Value
Reactor Core Power, MWth	2818
Peaking Factor	1.5
Number of Failed Fuel Rods	172
Fuel Decay Time, hr	24
Fraction of Core Inventory in Fuel Gap Kr-85 I-131 Other Noble Gases Other Iodines Alkali Metals	 0.10 0.08 0.05 0.05 0.12
Minimum Water Depth Above Damaged Fuel, ft	21
Iodine Decontamination Factor	142
Iodine Speciation in Fuel Gap, % Elemental Organic	 99.85 0.15
Fuel Activity Release Duration, hrs	2
Atmospheric Dispersion Factors	Tables 3.1.1 and 3.1.2

Table 3.2.3

Key Parameters Used in Radiological Consequence Analysis of Main Steam Line Break Accident

Parameter	Value
Reactor Core Power, MWth	2818
Failed Fuel, %	0
Reactor Coolant Activity, µCi/gm DE I-131 Equilibrium Iodine Activity Pre-accident Iodine Spike Activity	 0.2 2.0
Iodine-131 DCF, rem/Ci	3.29E+04
Iodine Speciation from Coolant, % Elemental Organic	 97 3
Time Until MSIV Isolation, sec	5.5
Coolant Mass Blowdown, lbm Liquid + Steam Mixture (7% quality) Steam Total	 3.9 E+04 1.4 E+04 5.3 E+04
Atmospheric Dispersion Factors	Tables 3.1.1 and 3.1.2

Table 3.2.4

Key Parameters Used in Radiological Consequence Analysis of Control Rod Drop Accident

Parameter	Value
Reactor Core Power, MWth	2818
Peaking Factor	1.5
Failed Fuel	
Cladding Failure, rods	1189
Melted, rods	11
Total Failure Fraction	0.0245
Fraction of Core Inventory in Fuel Gap	
Noble Gas	0.10
Iodine	0.10
Alkali Metals	0.12
Other Halogens	0.05
Iodine Speciation in Environment after Partitioning, %	
Elemental	97
Organic	3
Aerosol/Particulate	0
Isotopic Fractions Reaching the Turbine/Condenser	
Noble Gas	1.0
Iodine	0.1
Other Radionuclides	0.01
Isotopic Fractions Available for Environmental Release	
Noble Gas	1.0
Iodine	0.1
Other Radionuclides	0.01
Low Pressure (LP) Turbine/Condenser Volume, ft^3	172,000
LP Turbine/Condenser Leakage Rate, % per day	
0 – 24 hours	1.0
24 – 720 hours	0
LP Turbine/Condenser MVP Exhaust Rate, cfm	
0 – 24 hours	2200
24 – 720 hours	0
Atmospheric Dispersion Factors	Tables 3.1.1 and 3.1.2

Table 3.2.5

Key Parameters Used in Modeling the Control Room for Design Basis Radiological Consequence Analyses

Parameter	Value
Control Room Volume, ft^3	93,500
Normal Intake Rate, cfm	Not Used
Filtered Emergency Intake Rate, cfm 0 – 720 hours	 250
Recirculation Flow Rate, cfm 0 – 720 hours	 2100
Filter Efficiency, % Elemental Organic Aerosol/Particulate	 95 95 95
Unfiltered Inleakage Rate, cfm	115
Occupancy Factors 0 – 24 hours 24 – 96 hours 96 – 720 hours	 1.0 0.6 0.4
Ingress / Egress through TB	2 one-way trips per day, lasting 2 minutes each way
Breathing Rate, m^3/sec	3.5E-04
Atmospheric Dispersion Factors	Table 3.1.1

2.2 Electrical Engineering

The licensee's letters dated August 29, 2006, October 18, 2007, and March 13, 2008, were reviewed for this section of the safety evaluation.

2.2.1 Regulatory Evaluation – Electrical Engineering

The following NRC requirements are applicable to the NRC staff's review of the licensee's amendment request:

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.

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 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 50.67, "Accident Source Term," provides an optional provision for licensees to revise the AST used in design basis radiological analyses.

Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating DBAs at Nuclear Power Reactors," 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.

Regulatory Guide (RG) 1.75, Revision 3, "Criteria for Independence of Electrical Safety Systems," 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.

2.2.2 Technical Evaluation – Electrical Engineering

The HNP standby AC power supply consists of five diesel generators for both HNP units and supplies standby power for 4160 volt (V) emergency service buses 1E, 1F, 1G, 2E, 2F, and 2G. Diesel generators 1A, 1C, 2A, and 2C each supply an emergency bus while diesel generator 1B can supply either 4160-V emergency bus 1F or 2F.

Emergency buses 1E, 1F, and 1G are normally fed from startup transformer 1D with a backup feed from transformer 1C. The diesel generators cannot be paralleled with each other through the startup transformers bus supply breakers.

Each diesel generator unit consists of a diesel engine, generator, and associated auxiliaries mounted on a common base. Two completely independent air starting systems are furnished for each diesel engine either of which is capable of starting the diesel engine. Each of the air starting systems has adequate air capacity to start a single emergency diesel engine five times without recharging. Two motor-driven air compressors are available for each unit.

The diesel generators are housed in a reinforced concrete, Class 1 seismic structure which provides protection against natural phenomena such as tornado missiles, tornadoes, floods, lightning, rain, ice, or snow.

By letter dated August 29, 2006, the licensee proposed revising the HNP Unit 1 and 2 licensing basis with a full scope implementation of an AST. In their submittal, the licensee credited the HNP Unit 1 and 2 turbine building ventilation systems with purging the area around the main control room following a LOCA, CRDA, and a MSLB. The TB ventilation system is not designed as an ESF system.

The purpose of the TB ventilation system is to: (1) provide temperature control and air movement control for personnel comfort, (2) optimize performance by the removal of heat dissipated from plant equipment, (3) provide a sufficient quantity of filtered fresh air for personnel, (4) provide for air movement from areas of lesser potential airborne radioactivity to areas of greater potential airborne radioactivity prior to final exhaust, (5) minimize the possibility of exhaust air recirculation into the air intake, and (6) minimize the escape of potential airborne radioactivity to the outside atmosphere during normal operation by exhausting air, through a suitable filtration system, from the areas in which a significant potential for radioactive particulates and radioactive iodine contamination exist.

For each HNP unit, air is exhausted from the turbine building by a duct system to the outside environment via the reactor building vent plenum by two exhaust fans. The exhaust from the turbine building is filtered by two 50 percent (%) capacity filter trains. Each filter train consists of a bank of pre-filters, carbon absorbers, and high efficiency particulate air filters. Only one of the two 100% capacity exhaust fans is normally operating. If the operating exhaust fan fails, the standby exhaust fan starts automatically and an alarm is annunciated in the main control room.

The turbine building ventilation system incorporates redundancy and other features designed to assure turbine building operation for normal operation plant conditions. These features include a 100% standby supply air fan, a 100% standby exhaust air fan, two 50% capacity normally operating charcoal filter trains, and provision to adjust supply and exhaust fan flow rates manually so that one filter (50% of normal airflow) can be used during filter maintenance periods. In the event of a LOCA, CRDA, or MSLB, the licensee's operating procedures will ensure that turbine building exhaust ventilation (one of four turbine building exhaust fans) is initiated within 9 hours of the start of the accident. This time period is in accordance with the design basis assumptions for turbine building ventilation used in the HNP DBA analyses.

The licensee states that for the TB exhaust system to perform its post-accident function of purging the TB of radioactivity, one-of-four TB exhaust fans (two per HNP unit) must be able to operate, and the associated exhaust pathway must remain available. The pathway consists of ductwork and dampers. The licensee further states that any one of the four turbine building exhaust fans is capable of achieving the necessary minimum exhaust rate of 15,000 cubic feet per minute.

Loss of power to a single turbine building exhaust fan would result in a low flow annunciation in the main control room and the automatic start of the standby turbine building exhaust fan. Loss of power to both fans in one HNP unit would result in all forced turbine building exhaust flow being stopped if both exhaust fans in the other unit also failed.

The licensee states that no single failures exist that would impact the turbine building exhaust capability of both HNP units. The licensee further states that the only failure mechanism that could affect both HNP units is a seismic event. The air piping system supplies both safety and non-safety/non-seismic systems. A failure of the air systems of both HNP units would render both turbine building heating, ventilation, and air conditioning (HVAC) systems incapable of performing their required exhaust functions. Based on this potential consequence, the licensee provided a Regulatory Commitment to modify the HNP Unit 1 and 2 turbine building ventilation exhaust system to eliminate single point vulnerability to loss of system/instrument air. This issue has been included within the scope of a condition to the license as discussed below in Section 3.

In response to a NRC staff RAI, the licensee provided a Regulatory Commitment to modify the turbine building exhaust fans to provide the capability of supplying electrical power to the turbine building exhaust fans from an emergency power source. During normal plant operation the turbine building ventilation exhaust systems will not be connected to the Class 1E alternate power source. The licensee would only be required to connect the non-Class 1E turbine building exhaust fans to the Class 1E alternate power source during post-DBA operation provided offsite power cannot be restored within the required 9 hours. This issue has been included within the scope of a condition to the license as discussed below in section 3.

The licensee states that the power requirement for each fan is 40 horsepower (hp) for HNP Unit 1 and 75 hp for HNP Unit 2. In response to a staff request for additional information, the licensee demonstrated that the EDGs are able to support this additional loading.

The maximum allowable steady-state loading of diesel generator 1A for a LOCA with a loss of offsite power (LOOP) event during the 10 - 60 minute time frame for all cases of single failure is not expected to exceed 3250 kilo-watts (kW). The current maximum expected steady state loading on diesel generator 1A during the 10 - 60 minute time frame for a LOCA/LOOP event under conditions when another diesel generator or diesel battery has failed is 3192 kW. This is consistent with the HNP Unit 1 Final Safety Analysis Report (FSAR). The 40 hp TB exhaust fan motor will add 35 kW of load, considering fan motor efficiency, to this diesel. If this motor is started within the 10 - 60 minute window, the total load on diesel generator 1A will be 3227 kW (3192 kW plus 35 kW), which is less than the allowable loading of 3250 kW, the 168-hour rating.

The maximum allowable steady-state loading of diesel generator 2A for a LOCA with a LOOP event during the 10 - 60 minute time frame for all cases of single failure is not expected to exceed 3250 kW. The current maximum expected steady state loading on diesel generator 2A during the 10 - 60 minute time frame for a LOCA/LOOP event under conditions when another diesel generator or diesel battery has failed is 3164 kW. This is consistent with the HNP Unit 2 FSAR. The 75 hp turbine building fan motor will add 62 kW of load, considering fan motor efficiency, to this diesel. If this motor is started within the 10 - 60 minute window, the total load on diesel generator 2A will be 3226 kW (3164 kW plus 62 kW), which is less than the allowable loading of 3250 kW, the 300-hour rating.

The licensee states that the TB ventilation exhaust systems, specifically their AST credited function of purging the area around the main control room beginning 9 hours following the initiation of three of the four HNP DBAs, are not currently incorporated into the HNP Maintenance Rule program. In response to a staff request for additional information, the licensee states that the turbine building HVAC exhaust systems will be added to and evaluated in the HNP Maintenance Rule program consistent with 10 CFR 50.65. The NRC staff finds that this will provide further assurance that the systems will reliably perform their AST credited function. The licensee states that the HNP Maintenance Rule program evaluation will include consideration of the alternate safety related power supply and the associated manual transfer switches.

The licensee's existing TB ventilation system preventive maintenance procedure requires inspection and testing annually of each turbine building exhaust fan and associated dampers to ensure acceptable operation and condition. The licensee states that any abnormal conditions noted during the annual inspection would be corrected in accordance with plant corrective action programs.

As part of the implementation of the design change to provide a source of Class 1E power to the turbine building exhaust fans, the licensee will verify fan operation from the new power source. The licensee noted that the preventive maintenance and inspection tasks will be revised to periodically test operation of the fans from the Class 1E power source.

In response to a NRC staff RAI, the licensee states that the primary isolation device between the non-Class 1E turbine building exhaust fans and the Class 1E circuits will be a 150 ampere (A) safety related, environmentally and seismically qualified circuit breaker. For further protection and isolation, the licensee will use 150 amp (A) fuses that will be located in a seismically qualified manual transfer switch housing. The licensee evaluated the trip characteristics of the downstream protective devices (i.e., the new 150 A breaker and fuses) and determined that they are adequately coordinated with the upstream load center breaker over the entire range. The NRC staff finds that proposed breaker and fuse combination is consistent with the guidance provided in RG 1.75 and provides adequate protection and should prevent adverse effects of a fault to the rest of the distribution system.

The NRC staff also reviewed the environmental qualification portion of the license amendment request. The licensee used the methodology contained in TID 14844 to determine the radiation doses in the existing environmental qualification analyses. As mentioned 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, the environmental qualification of equipment should remain bounding during full-scope implementation of an AST.

2.2.3 License Conditions and Regulatory Commitments

Conditions to the Facility Operating License that relate to the provision of power supplies for the TB HVAC, and the air supply to the TB exhaust ventilation dampers, are discussed in Section 3.0 of this SE.

As part of the proposed AST amendment for the HNP, the licensee proposed three commitments in Enclosure 7 of its LAR. However, those items have been included within the license condition, as discussed above. In addition, the licensee provided a further commitment, as follows:

> The TB HVAC exhaust systems will be added to and evaluated in the HNP Maintenance Rule program consistent with 10 CFR 50.65 to provide reasonable assurance that the systems will reliably perform their AST credited function. This HNP Maintenance Rule program evaluation will include consideration of the alternate safety related power supply and the associated manual transfer switches. The inclusion of TB HVAC exhaust systems in the HNP Maintenance Rule program will be complete by May 31, 2010.

2.2.4 Summary

Based on the evaluation and implementation of the License Conditions and Regulatory Commitments listed above, the NRC staff finds the proposed revision to the HNP licensing basis provides reasonable assurance that an acceptable power supply will be implemented at HNP Units 1 and 2 for the TB ventilation exhaust system. The NRC staff also concludes that the proposed changes are in accordance with 10 CFR 50.49, 10 CFR 50.65, 10 CFR 50.67, and the requirements of GDCs 17 and 18. Therefore, the NRC staff finds these changes, as discussed in Section 2.2, to be acceptable.

2.3 Mechanical and Civil Engineering

2.3.1 Introduction

Systems and equipment needed to mitigate DBAs are required to be designed and constructed to Seismic Category I criteria in order to withstand earthquakes. Since portions of the HNP Unit 1 main steam piping, drain lines, and main condenser that are located outside of the PC are not specifically designed to Seismic Category I criteria for withstanding the effects of earthquakes, the licensee utilized the approach presented in the NRC approved GE topical report NEDC-31858P-A, “BWROG Report for Increasing MSIV Leakage Rates and Elimination of Leakage Control Systems,” (Reference 19) to demonstrate that the alternate leakage treatment (ALT) pathway is seismically rugged to satisfy the functional requirements for equipment affected by the license amendment request (LAR). The approved version of the topical report was issued in

August 1999, and a copy of the NRC staff's safety evaluation of the report, which had been issued on March 3, 1999, is included in the published approved version of the topical report. The report, including the NRC staff's SE, will be referred to as the "approved BWROG topical report."

The HNP ALT pathway for Unit 1 is reviewed in this SE. The HNP ALT pathway for Unit 2 was reviewed in Reference 25. The HNP ALT pathway for Unit 1 utilizes the large volume of the main steam lines (MSLs) and the main condenser to provide holdup and plate-out of fission products that may leak through the closed main steam isolation valves (MSIVs). The primary components of the ALT pathway are the main condenser, the MSLs from the MSIVs to the turbine stop and bypass valves, and the drain piping which originates downstream of the outboard MSIVs and terminates at the main condenser. The condenser forms the ultimate boundary of the ALT pathway. In addition, as part of a full scope implementation of Alternative Source Term (AST), deposition in the main condenser is credited for those secondary containment bypass leakage paths that terminate in the main condenser. Furthermore, The Units 1 and 2 turbine building ventilation exhaust systems are credited in AST with purging the area around the main control room following a loss-of-coolant accident, main steam line break accident, and CRDA.

The methods and criteria used to provide the generic basis for acceptability of individual licensee applications are described in the approved BWROG topical report. The NRC staff found in its SE on the topical report that the report's approach of utilizing the earthquake experience-based methodology, supplemented by plant-specific seismic verification walkdowns and analytical evaluations, is an acceptable basis to demonstrate the seismic ruggedness of non-seismically analyzed ALT pathway (e.g., main condensers and related piping, and turbine building). The approved BWROG report summarizes data on the seismic performance of the main condensers and related piping, and turbine buildings in past strong-motion earthquakes at various facilities, and compares design attributes of the main condensers and turbine buildings with those in typical GE Mark I, II, and III nuclear plants. However, the approved BWROG topical report identified certain limitations that required individual licensees to provide plant-specific design information and evaluation when the approved BWROG topical report approach was elected for resolving the MSIV leakage issue.

The approved BWROG topical report was referenced by the licensee as the bases for the acceptability of the proposed LAR changes. The licensee's August 29, 2006 LAR included Enclosure 8, "Edwin I. Hatch Nuclear Plant Unit 1 Main Steam Isolation Valve Alternate Leakage Treatment Path Description and Seismic Evaluation." That enclosure includes the HNP Unit 1 seismic ruggedness verification of the ALT path in accordance with the recommendations and requirements of Reference 19.

In accordance with Regulatory Guide (RG) 1.183, holdup and deposition in the main condenser of any primary containment leakage that bypasses the secondary containment and is routed to the main condenser may be credited in the analysis. Such crediting of deposition for bypass leakage is allowed on a case-by-case basis per section 4.5 of RG 1.183 Appendix A. LAR Enclosure 9, entitled "Edwin I. Hatch Nuclear Plant Unit 1 Seismic Verification of Potential Secondary Containment Bypass Leakage Paths Terminating at the Main Condenser" and LAR Enclosure 10, entitled "Edwin I. Hatch Nuclear Plant Unit 2 Seismic Verification of Potential Secondary Containment Bypass Leakage Paths Terminating at the Main Condenser" were included with the licensee's August 29, 2006 LAR. These enclosures document the seismic adequacy review of HNP Units 1 and 2 bypass leakage piping and supports in accordance with the recommendations and requirements of Reference 19.

The licensee's August 29, 2006, LAR included Enclosure 11, "Hatch Nuclear Plant Unit 1 Seismic Verification of the Turbine Building Exhaust Ductwork" and Enclosure 12, "Hatch Nuclear Plant Unit 2 Seismic Verification of the Turbine Building Exhaust Ductwork" which document the seismic verification of the Units 1 and 2 turbine building exhaust ductwork.

The licensee supplemented its application with additional information in a letter dated July 16, 2007 (Reference 6), December 11, 2007 (Reference 9) and February 4, 2008 (Reference 11) in response to the NRC staff's RAI.

2.3.2 Mechanical and Civil Engineering – Regulatory Evaluation

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.67 "Accident Source Term," provides a mechanism for licensed power reactors to replace the traditional accident source term used in their DBA analyses with an AST.

Regulatory guidance for the implementation of an AST is provided in RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors". 10 CFR 50.67 requires a licensee seeking to use an AST to apply for a license amendment and requires that the application contain an evaluation of the consequences of affected DBAs.

For Boiling Water Reactor (BWR) MSIV leakage, RG 1.183 allows credit for reducing MSIV releases due to holdup and deposition in the main steam piping downstream of the MSIVs and in the main condenser, including the treatment of air ejector effluent by off-gas systems, if the components and piping systems used in the release path are capable of performing their safety functions during and following a Safe Shutdown Earthquake (SSE).

2.3.3 Mechanical and Civil Engineering - Technical Evaluation

The NRC staff reviewed the Unit 1 MSIV ALT pathway seismic evaluation (LAR Enclosure 8), the Units 1 and 2 seismic verification of secondary containment bypass leakage paths terminating at the main condenser (LAR Enclosures 9 and 10), and Units 1 and 2 seismic verification of the turbine building exhaust ductwork (LAR Enclosures 11 and 12). The methodology for evaluating the seismic ruggedness of the MSIV ALT pathway is provided in the approved BWROG topical report. Given the similarities of the MSIV ALT pathway and the primary containment leakage that bypasses the secondary containment and is routed to the main condenser, the licensee evaluated the seismic ruggedness of the secondary containment bypass piping in accordance with the BWROG topical report and its associated NRC SE. The NRC staff finds this methodology acceptable.

The NRC staff determined that additional information would be required from the licensee in order to facilitate the review the licensee's response to these requests was provided in letters dated July 16, and December 11, 2007, and February 4, 2008.

Several of the following sections refer only to Unit 1. As discussed further in Section 2.3.3.7

below, the Unit 2 main condenser, MSIV ALT piping and pipe supports are already credited for holdup and deposition of MSIV leakage in accordance with the approved BWROG topical report (see References 20, 21 and 22).

2.3.3.1 <u>Seismic Ground Motion</u>

In the SE included with Reference 19, the NRC staff reviewed and accepted seismic ground motion of 13 selected sites in the earthquake experience database included in the approved BWROG topical report. From these 13 database sites, SNC selected nine facilities to compare the database site spectra with the HNP Design Basis Earthquake (DBE). As shown in Figure 2 of LAR Enclosure 8 and Figure 3-1 of LAR Enclosures 9 and 10, the 5 percent (%) damped HNP DBE and ½ Seismic Margin Earthquake (SME) are enveloped by the selected database site spectra in the frequency band above 1 Hz.

The full SME in-structure response spectra (ISRS) are for ground motion response spectra with a NUREG/CR-0098 median centered ground motion with a peak ground acceleration (PGA) of 0.3g. The ISRS accepted for resolution of USI A- 46 for plant Hatch is the ½ SME ISRS. For bounding seismic piping analytical evaluations, the licensee employed the turbine building ½ SME ISRS. The HNP DBE is enveloped by the ½ SME ground spectrum.

Considering the above, the NRC staff determines that it is appropriate to use the earthquake experience database approach for demonstrating seismic ruggedness of the Unit 1 non-seismically analyzed ALT pathway (e.g., main condensers and related piping, and turbine building) and Units 1 and 2 secondary containment bypass leakage paths terminating at the main condenser as well as to utilize the turbine building ½ SME ISRS for bounding piping system evaluations.

2.3.3.2 <u>Unit 1 Main Condenser</u>

SNC performed a seismic verification walkdown and reviewed condenser design documents to ensure that the HNP Unit 1 main condensers fall within the design parameters (design codes, overall dimensions, condenser shell thickness and material properties, heat transfer area, anchorage configuration, etc.) of the earthquake experience database contained in Appendix D of the BWROG topical report.

Based on the information in LAR Enclosure 8 and response to the NRC staff RAI (References 6 and 11), each condenser is supported on four 2-inch thick support feet (sole plates) which in turn are supported on four reinforced concrete piers. At each sole plate, there are 4 - 2¼" diameter ASTM A36 cast-in-place anchor bolts. The sole plates are slotted radially to allow thermal movement of the condenser. Each sole plate has 1 ½" thick shear lugs that extend 4" into the reinforced concrete pier. There is a single shear plate, 50" long, running in the E-W direction and located along the center line of the sole plate. There are four shear plates, each 14.25" long, running in the N-S direction and located on either side of the single E-W shear plate. The condenser piers are supported on the turbine building base mat, are approximately 4 feet high and have plan dimensions of 4'-10" by 3'-2", and are reinforced by 30 #9 vertical reinforcing steel and 4 sets of horizontal #5 ties spaced 12" on center. The condenser base shear is transferred to the anchor bolts, to the sole plates/shear lugs, and then to the concrete pier and to the turbine building base mat. In addition to the anchor bolts at the sole plates, there are a total of 16 anchor bolts at two sides of the main condenser which were installed to compensate for increased uplift loads for net reduction in weight due to change in tube material. These side anchor bolts only resist uplift.

The licensee states that the anchorage for the HNP Unit 1 condenser is comparable with the anchorages for similar condensers in the earthquake experience database. The shear areas of the condenser anchorage, in the directions parallel and transverse to the turbine generator axis, divided by the seismic demand, were used to compare with those presented in the BWROG topical report 31858. The HNP Unit 1 condenser anchorage shear area to seismic demand is greater than those in the selected database sites as depicted in Figures 5a and 5b of Reference 1, Enclosure 8.

In responses to the NRC staff RAI (References 6 and 11), the licensee states that the condenser piers have significant capacity compared to the calculated demand. The maximum bending and shear stresses in the shear lug and in the weld between the shear lug and the sole plate were also evaluated and found acceptable. The transfer of load from shear lug to the concrete was checked and found that the capacity exceeded the maximum demand.

Additionally, in response to a NRC staff RAI (Reference 11), the licensee states that the results of an evaluation of the condenser shell and structural elements at the four corner supports showed that the capacity of the structural elements exceeded the maximum demand.

Based on review of the information provided by the licensee, the NRC staff finds the licensee's approach to demonstrate seismic adequacy of the main condenser and associated anchorages reasonable and acceptable.

2.3.3.3 Unit 1 Turbine Building

Performance of the turbine building during or after a design basis seismic event is of interest to the issue of MSIV leakage only to the extent that the structure and components should survive and not degrade the capabilities of the selected main steam and condenser pathways.
The licensee states in LAR Enclosure 8 that the HNP Unit 1 turbine building superstructure (above the turbine floor) consists of moment-resisting structural steel frames in the East-West direction, and braced frames in the North-South direction. The turbine building substructure (below turbine floor elevation) consists of reinforced concrete shear walls with pilasters that support the steel superstructure. The reinforced concrete shear walls resist the lateral loads in N-S and E-W directions.

As stated in LAR Enclosure 8 and as noted in the HNP FSAR, the turbine building is classified as a Category II structure. Furthermore, the licensee states that the turbine building was designed for 300 mile per hour (mph) tornado wind loading, tornado missile impact effect, and tornado induced differential pressure. The structural elements were designed to maintain the resulting stresses below yield for the tornado load combination.

The licensee also states in LAR Enclosure 8 that, as part of seismic margin assessment (SMA) of HNP Unit 1, the turbine building was evaluated to have a high confidence of low probability of failure (HCLPF) of 0.3g peak ground acceleration which is twice the amplitude of the HNP Unit 1

DBE. (Note: The NRC staff previously reviewed the HNP Unit 1 SMA and found it acceptable as documented in Reference 27).

The licensee provided further information in response to the NRC staff RAI (Reference 6) relative to turbine building maximum seismic story shears corresponding to ½ SME. The licensee, as stated in Reference 6, evaluated the turbine building structural elements (reinforced concrete shear walls and pilasters) where seismic shear force exceeded the shear force due to the tornado loading and concluded that in all cases the design code allowable limits will be maintained within the allowable capacity to demand ratio of at least 2.2.

Based on review of the information presented by the licensee, the NRC staff finds the licensee's approach to demonstrate seismic adequacy of the HNP Unit 1 turbine building reasonable and acceptable.

2.3.3.4 Unit 1 Turbine Pedestal

As noted in LAR Enclosure 8 and as described in SNC's response to the NRC staff RAI (Reference 11), the turbine pedestal is a separate reinforced concrete structure housed within the turbine building. The turbine pedestal is supported on the turbine building basemat but it is not connected to turbine building above the basemat. There is a one inch gap between the turbine pedestal deck and the adjacent turbine building floor. In Reference 11, the licensee states that the turbine pedestal and components attached to the turbine pedestal were evaluated by the licensee's seismic review team (SRT) and the SRT found that there was no potential seismic interaction with the condenser or the MSIV ALT path piping. The bases for this conclusion are as follows:

a. The seismic analysis of the turbine building for ½ SME showed that the absolute sum of relative displacements of the turbine pedestal and turbine building were less than the existing one inch gap between the turbine pedestal deck and the adjacent turbine building floor.

b The bellows connecting the condenser and the turbine can accommodate the ½ SME displacement demand.

c The turbine pedestal is a massive stiff reinforced concrete structure that was originally designed to satisfy the turbine generator manufacturer's static and dynamic displacement criteria under various operating and accident loading conditions.

d The SRT found the components connected to the turbine pedestal to be well anchored.

Based on review of the information presented by the licensee, the NRC staff finds the licensee's response to the RAI relative to the turbine pedestal potential seismic II/I concern reasonable and acceptable.

2.3.3.5 Unit 1 Alternate Leakage Treatment (ALT) Path

To confirm the capability of the main steam piping to serve as part of an ALT path, the Licensee performed seismic verification walkdowns to ensure that the main steam and turbine bypass piping, the steam drain lines, and the interconnecting piping that were not seismically analyzed fall within the bounds of the design characteristics of the seismic experience database of Reference 19. LAR Enclosure 8, summarized the seismic ruggedness verification of the piping system and the associated supports. This report states that the HNP Unit 1 seismic verification of the ALT path was performed in accordance with the recommendations of the approved BWROG topical report.

As stated in LAR Enclosure 8, portions of the Unit 1 HNP main steam and drain piping systems were originally seismically analyzed. The analyzed lines included the main steam piping (from the MSIV to the turbine stop valves), the turbine bypass (to the bypass valves), the drain line portion in the reactor building, and the portions of various main steam branch connections to the seismic anchors downstream of the isolation valves. Design methods for these analyzed lines are consistent with Seismic Category I analysis methods for HNP Unit 1.

According to the licensee, the remainder of the HNP Unit 1 main steam system piping, including main steam drain to the condenser and interconnected systems, is made of welded steel piping and standard support components, and was designed by rule and approximate methods. This piping is similar in diameter, thickness, and material to those installed in the plants that are in the earthquake experience database.

Piping

The Unit 1 MSIV ALT pathway description, scope of review and seismic evaluation is included in LAR, Enclosure 8. Per the BWROG topical report (Reference 19), primary components to be relied upon for pressure boundary integrity in resolution of the BWR MSIV leakage issue are: (1) the main turbine condenser (see section 2.3.3.2 above), (2) the main steam lines from the MSIVs to the turbine stop and bypass valves, and (3) the main steam turbine bypass and/or drain line piping to the condenser. The turbine bypass lines to the main condenser or the main steam drain lines to the main condenser can be utilized to transfer leakage from the MSIVs to the main condenser. In SNCs response to the NRC staff RAI, SNC states that they chose to use the drain line path over the turbine bypass line path because it would be much more difficult to assure the turbine bypass valves could be opened following a loss of offsite power to establish an MSIV leakage pathway. This is consistent with the approved BWROG topical report.

SNC has submitted information on a detailed seismic analysis in the current design basis (CDB) of the seismic adequacy of the main steam and turbine bypass lines, utilizing the response spectrum method in accordance with the ANSI B31.7 Nuclear Power Piping Code, Class 2, 1969 edition (ANSI B31.7). The analysis model included the main steam line (to the turbine), the bypass line, and branch piping (moisture separator reheater, steam jet air ejector, and the reactor feed pump branches) up to the first anchor downstream of the isolation valve. The main steam drain line in the reactor building was also analyzed seismically in the CDB. The remainder of the pipe lines in the ALT pathway including the main steam drain to the main condenser and process steam lines which interconnect with the main steam line between the

MSIVs and turbine stop valves are seismically evaluated in the LAR in accordance with the generic method of the approved BWROG topical report. The BWROG approach of verifying the seismic adequacy of the ALT piping is based on utilizing the earthquake experience-based methodology, supplemented by a plant-specific walkdown and performing a bounding seismic analytical evaluation. The licensee walked down all piping included in the ALT pathway and included the currently seismically analyzed portions in the seismic verification walkdown to ensure that there were no seismic vulnerabilities not considered in the analysis.

LAR Enclosure 8, section 3 and Table 1, contain summaries of piping data (sizes, schedules, materials, etc) from the ALT pathway walkdown piping. The NRC staff performed a comparison of the HNP Unit 1 walkdown piping data with the same data for the facilities in the earthquake experience database contained in the approved BWROG topical report. This comparison showed that pipe sizes and ratios (D/t) of pipe diameter (D) to pipe wall nominal thickness (t) for the HNP ALT pathway and associated boundary piping are for the most part well-represented within the pipe sizes and ratios of the earthquake experience database piping. However, there are exceptions where some of the walkdown piping sections are not explicitly represented in the earthquake experience database. Considering that piping of both smaller and larger size with comparable and enveloping D/t ratios are adequately represented in the database, the NRC staff finds that the HNP Unit 1 ALT pathway piping is adequately represented in the BWROG topical report earthquake experience database.

The walkdowns identified vulnerabilities that included inadequately supported piping, seismic interactions, pipe support and anchorage deficiencies. These vulnerabilities were identified as outliers and are listed in LAR Enclosure 8, Table 2. The identified outliers were resolved either by analysis or by modification. SNC states in their December 11, 2007, response to RAI 18, that all modifications listed in Table 2 were completed as part of a design change package implemented during the Unit 1 outage 1R22 which ended on April 1, 2006. Modifications included adding supports, rerouting piping, replacing and modifying existing supports and modifying walls to allow pipe motion without interaction.

In order for the NRC staff to conclude that the proposed ALT piping system will maintain its functionality under the plant design basis SSE, Section 5.8 of the NRC staff's SE on the approved BWROG topical report requires that the licensee provide for staff review a summary of the bounding seismic analysis for a represented portion of the ALT drain piping. The licensee's bounding seismic analysis was performed on the three inch drain line to condenser, starting at the reactor building and turbine building interface anchor. The portion of the main steam drain line inside the reactor building is seismically analyzed in the CDB using the response spectrum method. The main steam 3-inch drain line is the ALT primary drain path to the condenser and is the largest drain line in the scope of the MSIV ALT pathway. Therefore, the licensee considered the 3-inch main steam line drain in the turbine building for the bounding seismic analysis of the main steam drain piping. The seismic analysis was performed utilizing the response spectrum method (see SE Section 2.3.3.1 above for spectrum discussion). The licensee evaluation summary is contained in Section 3 of LAR Enclosure 8. The results of the piping stress analysis yielded safety factors of 1.72, 1.96, and 1.78 relative to the code limits for sustained loads, faulted occasional loads, and thermal expansion loads, respectively. The NRC staff finds the licensee's bounding seismic analysis acceptable as it is in accordance with the approved BWROG topical report. The NRC staff inquired about the combined maximum displacement that resulted from the bounding analysis and the clearance used during the walkdowns to evaluate seismic interactions. In its response to the NRC staff RAI, the licensee states that the maximum seismic displacement was less than 6 inches and 6 inches was used for seismic interaction screening purposes during the seismic walkdowns along with the judgment of the seismic capability engineers performing the walkdowns. The NRC staff finds the licensee's response acceptable.

The licensee has used seismically analyzed piping in the CDB and has utilized the approved BWROG topical report method provisions and limitations for non-seismically analyzed piping to demonstrate the seismic ruggedness of the HNP Unit 1 MSIV ALT pathway piping. Therefore, based on its review as summarized above, the NRC staff finds that the HNP Unit 1 MSIV ALT pathway piping is seismically rugged for the proposed AST.

Pipe Supports

As stated in the LAR Enclosure 8, all pipe supports for the main drain to condenser inside the turbine building were evaluated. For the portion of the main drain inside the turbine building, the licensee determined the safety margin to be at least four times the Hatch DBE. LAR Enclosure 8 also stated that the interconnected systems consist of standard support components that are well-represented in the earthquake experience data base. The licensee evaluated the most heavily loaded supports for the interconnected systems to determine the safety margin in the pipe support and the anchorage design. The licensee determined that a safety margin of at least three times the Hatch DBE exists.

Considering the acceptable resolution of the identified outliers by either detailed evaluations or modifications and that all selected pipe supports were shown to have capacities larger than the plant-specific seismic demand, the NRC staff concludes that there is reasonable assurance that the HNP Unit 1 alternate leakage path pipe supports are seismically adequate for the intended purpose.

2.3.3.6 Unit 1 Secondary Containment Bypass Leakage Paths Terminating at the Main Condenser

The bypass leakage piping consists of the HPCI steam drain (1 inch diameter), RCIC steam drain (1 inch diameter) and RWCU blow-down line to the condenser (4 inch diameter). The majority of the piping is located in the Unit 1 turbine building with the exception of the RCIC line which extends into the control building.

LAR, Enclosure 9, entitled "Edwin I. Hatch Nuclear Plant Unit 1 Seismic Verification of Potential Secondary Containment Bypass Leakage Paths Terminating at the Main Condenser" summarized the seismic ruggedness verification of the piping system and the associated supports. This report states that the seismic verification was performed in accordance with the recommendations of the approved BWROG topical report.

The seismic review team consisted of two senior SNC personnel, each with minimum of 20 years experience in seismic capability evaluation and nuclear seismic design. The SRT performed a seismic verification walkdown of the secondary containment bypass leakage piping and its associated pipe supports to ensure that the design attributes are consistent with industry standard practices and to screen for known seismic vulnerabilities.

Piping

The HNP Unit 1 secondary containment bypass leakage paths description, scope of review and seismic evaluation is contained in LAR Enclosure 9.

In order to credit deposition for bypass leakage in the main condenser, it is assumed that the piping routing the bypass leakage to the main condenser is capable of performing its required function during and after a safe shutdown earthquake. Such crediting of deposition for bypass leakage is allowed on a case-by-case basis per section 4.5 of RG 1.183, Appendix A. Therefore, it is necessary to demonstrate that the bypass leakage piping is seismically rugged to perform its intended function.

The seismic verification of the non-seismically analyzed piping was performed in accordance with the approved BWROG topical report generic method. LAR Enclosure 9, Section 3, contains a summary comparison which shows that both pipe sizes and diameter to thickness (D/t) ratios for the bypass leakage piping fall within the limits of the pipe sizes and D/t ratios for the earthquake experience database piping. In addition, the material summary comparison found that the bypass leakage pipe materials are comparable with piping within the earthquake experience database. In LAR Enclosure 9, Figure 3-1, SNC provided a comparison which shows that the Hatch 1/2 SME Ground Spectrum is generally bounded by those of the earthquake experience database sites. Therefore, the NRC staff finds the approved BWROG topical report generic method acceptable for verifying the seismic ruggedness of non-seismically analyzed secondary containment bypass leakage piping of Unit 1.

The licensee walked down all the in-scope Unit 1 bypass leakage piping and included the seismic walkdown evaluation and results in LAR Enclosure 9. Seismic movements of up to 6 inches in any horizontal direction were considered for seismic interaction. The walkdown evaluation concluded that all piping is adequately supported with piping spans, generally, in accordance with requirements for B31.1 deadweight spans. There were four outliers identified during the SRT walkdown. One support for the high pressure coolant injection/reactor core isolation cooling (HPCI/RCIC) piping had physical damage (one leg of the angle support had a notch). The licensee evaluated this outlier and found it acceptable. In addition, the licensee identified three potential interaction conditions, two for reactor water cleanup (RWCU) (interaction with trapeze support and insulation for a pipe located above RWCU) and one for RCIC piping (concrete masonry unit (CMU) wall penetration). The licensee evaluated the RWCU interaction conditions and found them acceptable. The licensee also verified that the CMU wall which the RCIC pipe penetrates was examined during the walkdown, evaluated and seismically qualified as part of IE Bulletin 80-11 program on masonry wall design; thus the outlier for the RCIC piping was found acceptable. The licensee's outlier resolution is contained in Section 7.1 of LAR Enclosure 9. The NRC staff finds the licensee's outlier resolution acceptable, as it employs good engineering judgment along with plant design basis conformance.

For the bounding seismic analytical assessment, the licensee compared the Unit 1 bypass piping with the Unit 2 bypass piping. As in Unit 2, the Unit 1 HPCI and RCIC piping run together on common supports and are generally rigidly supported with U-bolts providing lateral and vertical support. The Unit 1 and Unit 2 RWCU piping systems have similar routing and differences are not significant. For both units, the RWCU piping is flexibly supported on vertical rod hanger supports and subject to large displacements under seismic loading. Therefore, the RWCU piping was selected for a bounding seismic analysis to provide assurance of the bypass leakage piping seismic ruggedness. Selecting the Unit 2 RWCU piping model for bounding seismic analytical assessment for both units, the licensee's contractor, ABS Consulting/EQE International, performed seismic analysis utilizing the response spectrum method. LAR Enclosure 10, Section 8, contains a summary of the analytical evaluation. Review of the analytical evaluation summary shows that the piping system satisfies the acceptance criteria with a ratio of actual to allowable stress of less than 0.3. In addition, the licensee noted that the Unit 1 piping system is better supported than that of Unit 2. Based on the above review and the NRC staff's review of the Unit 2 bounding analytical evaluation as presented in Section 2.3.3.4.3.1 of this SE, the NRC staff finds the licensee's Unit 1 limited analytical assessment acceptable.

The licensee has employed the approved BWROG topical report method to demonstrate the seismic ruggedness for the non-seismically analyzed piping of the HNP Unit 1 secondary containment bypass leakage piping. Therefore, based on its review as summarized above, the NRC staff concludes that the HNP Unit 1 bypass leakage piping is seismically rugged for the proposed AST.

Pipe Supports

The pipe supports were in good condition with no evidence of excessive corrosion. The licensee performed a plant-specific seismic evaluation of six representative pipe supports (four supports for RCIC and HPCI, and two supports for RWCU piping) to ensure seismic adequacy of the pipe support system. A conservative factor of 1.25, in accordance with SQUG GIP-2 (Reference 23) Section 4.4.3, was used to increase the seismic demand for the evaluation of the pipe supports. The concrete expansion anchors were evaluated in accordance with Appendix C of SQUG GIP-2. The results of the evaluations showed that the pipe supports for the Unit 1 HNP bypass leakage piping are adequate for the seismic demand of ½ SME which envelops the Unit 1 DBE.

Considering the acceptable resolution of the identified outliers and that all six selected pipe supports were shown to have capacities larger than the plant-specific seismic demand, the NRC staff concludes that there is reasonable assurance that the pipe supports for the Unit 1 HNP secondary containment bypass leakage piping are seismically adequate for the intended purpose.

2.3.3.7 Unit 2 Secondary Containment Bypass Leakage Paths Terminating at the Main Condenser

The licensee contracted ABS Consulting to conduct the seismic ruggedness verification for Unit 2 secondary containment bypass leakage paths. Report No. 1302241-R-002 entitled "Hatch Nuclear Plant Unit 2 Seismic Verification of Potential Secondary Containment Bypass Leakage Paths Terminating at the Main Condenser" was attached as Enclosure 10 to the LAR. This report states that the seismic verification was performed in accordance with the recommendations of the approved BWROG topical report. As stated in LAR Enclosure 10, the entire secondary containment bypass leakage piping is located in the Unit 2 turbine building and consists of the HPCI steam drain (1 inch diameter), RCIC steam drain (1 inch diameter) and RWCU blowdown line to the condenser (4 inch diameter).

The seismic review team (SRT) consisted of two senior ABS Consulting personnel, each with more than 30 years experience in seismic capability evaluation and nuclear seismic design. The SRT performed a seismic verification walkdown of the secondary containment bypass leakage piping and its associated pipe supports to ensure that the design attributes are consistent with industry standard practices and to screen for known seismic vulnerabilities.

Piping

LAR Enclosure 10 includes the Unit 2 secondary containment bypass leakage paths description, scope of review and seismic evaluation.

In accordance with the HNP Unit 2 current licensing basis, the main condenser is already credited for holdup and deposition of MSIV leakage in accordance with the approved BWROG topical report (see References 20, 21 and 22). In order to credit deposition for bypass leakage in the main condenser, it is assumed that the piping routing the bypass leakage to the main condenser is capable of performing its required function during and after a safe shutdown earthquake. Such crediting of deposition for bypass leakage is allowed on a case-by-case basis per section 4.5 of RG 1.183 Appendix A. Therefore, it is necessary to demonstrate that the bypass leakage piping is seismically rugged to perform its intended function.

The seismic verification of the non-seismically analyzed piping was performed in accordance with the approved BWROG topical report generic method. LAR Enclosure 10, Section 3 contains a summary comparison which shows that both pipe sizes and D/t ratios for the bypass leakage piping fall within the limits of the pipe sizes and D/t ratios for the earthquake experience database piping. In addition, a material summary comparison found that the bypass leakage pipe materials are comparable with piping within the earthquake experience database. In Figure 3-1 of LAR Enclosure 10, the licensee provided a comparison which shows that the HNP 1/2 SME Ground Spectrum is generally bounded by those of the earthquake experience database sites. Therefore, the NRC staff finds the method of the approved BWROG topical report generic method acceptable for verifying the seismic ruggedness of non-seismically analyzed secondary containment bypass leakage piping of Unit 2.

The licensee walked down all the in-scope Unit 2 bypass leakage piping and included in LAR Enclosure 10 the walkdown data sheets, as well as the seismic walkdown evaluation and results. Seismic movements in any horizontal direction of up to 6 inches were considered for seismic interaction. The walkdown evaluation concluded that all piping is adequately supported with piping spans, generally, in accordance with requirements for the B31.1 code deadweight spans. There were no outliers identified from the walkdowns and all piping and supports were found to meet the walkdown criteria.

The HPCI and RCIC piping run together on common supports and are generally rigidly supported with U-bolts providing lateral and vertical support. The RWCU piping is flexibly supported on vertical rod hanger supports and subject to large displacements under seismic loading. Therefore, the RWCU piping was selected for a bounding seismic analysis to provide assurance of the bypass leakage piping seismic ruggedness. The seismic analysis was performed utilizing the response spectrum method (see SE section 2.3.3.1 for spectrum discussion). Section 8 of Enclosure 10 contains a summary of the analytical evaluation. Review of the analytical evaluation summary shows that the piping system satisfies the acceptance criteria. Maximum combined stress due to occasional loads for faulted condition resulted in a ratio of calculated-to-code-allowable of 0.29, which yields a safety factor of 3.4. The maximum displacement is 3 inches, which is less than the displacement of 6 inches that was used during the walkdowns to evaluate potential seismic proximity interactions. The NRC staff finds the licensee's bounding seismic analysis acceptable as it is in accordance with the approved BWROG topical report.

The licensee has employed the approved BWROG topical report method to demonstrate the seismic ruggedness for the non-seismically analyzed piping of the HNP Unit 2 secondary containment bypass leakage piping. Therefore, based on its review as summarized above, the NRC staff finds that the HNP Unit 2 secondary containment bypass leakage piping is seismically rugged for the proposed AST.

Pipe Supports

The pipe supports were in good condition with no evidence of excessive corrosion or physical damage. There were no outliers identified during the SRT walkdown.

The licensee performed a plant-specific seismic evaluation of all supports for RCIC, HPCI and RWCU piping (in the scope of this evaluation) to ensure seismic adequacy of the support systems. A conservative factor of 1.25, in accordance with SQUG GIP-2 Section 4.4.3, was used to increase the seismic demand for the evaluation of the pipe supports. The concrete expansion anchors were evaluated in accordance with Appendix C of GIP-2. The results of the evaluations showed that the pipe supports for the HNP Unit 2 bypass leakage piping are adequate for the seismic demand of ½ SME which envelops the Unit 2 DBE.

Considering that the SRT walkdowns did not identify any outlier and all pipe supports were shown to have capacities larger than the plant-specific seismic demand, the NRC staff concludes there is reasonable assurance that the pipe supports for the HNP Unit 2 secondary containment bypass leakage piping are seismically adequate for the intended purpose.

2.3.3.8 Turbine Building Exhaust Ductwork

The HNP TB ventilation systems are credited in the AST analysis with purging the area around the main control room (MCR) for the removal of activity at an exhaust rate of 15,000 cubic feet per minute (cfm) following a loss-of-coolant accident (LOCA), a CRDA , and a main steam line break (MSLB) accident. LAR Enclosures 11 and 12 contain the description, scope of review and seismic verification of the TB exhaust ductwork for HNP Units 1 and 2, respectively.

The licensee's evaluation methodology for the seismic ruggedness of the TB ventilation systems follows the guidelines of LAR Enclosure 13, which is the Electric Power Research Institute (EPRI) Report No.1007896, "Seismic Evaluation Guidelines for HVAC Duct and Damper Systems", supplemented by the peer review comments contained in LAR Enclosure 14. The EPRI guidelines are based on seismic experience data. The methodology is similar to GIP-2, Reference 23, and its associated NRC SE, Reference 24. It relies on the evaluation of seismic failure mechanisms for duct and damper systems from seismic experience data and includes methods to screen and identify the ductwork seismic vulnerabilities and weaknesses. Similarly to GIP-2, the EPRI methodology includes performing in-plant walkdowns and reviews to determine the seismic adequacy of the duct systems and supports, selecting representative duct and support analytical review samples and performing analytical reviews, and resolving outliers. The EPRI guidelines also include training and experience qualification requirements for individuals performing the seismic reviews similar to those of GIP-2. The EPRI Report No.1007896 has not been approved by the NRC staff for use beyond this review. However, based on a preliminary review to date, the NRC staff has found the EPRI Report No.1007896 approach of utilizing the earthquake experience-based methodology as supplemented by the peer review comments and by plant-specific seismic ruggedness evaluations, to be an acceptable basis to demonstrate the seismic ruggedness of non-seismically analyzed ventilation systems that are credited in the proposed HNP AST analysis. The NRC staff's acceptance here of utilizing the seismic experience-based methodology of the above referenced EPRI report is limited to its application to the proposed HNP AST amendment, as discussed in this SE, and is not an endorsement for its use for other applications at HNP Unit 1 and Unit 2.

The NRC staff reviewed the above enclosures and the licensee's responses to NRC staff RAIs for the seismic ruggedness of the HNP TB ventilation systems in the scope of review, as credited in the proposed AST analysis. The TB ventilation systems consist of a supply system and an exhaust system, but only the exhaust systems are credited in the AST analyses for purging the TB. The extent of the licensee's review included ductwork, duct supports and associated in-line components such as registers, dampers, damper actuators, in-line fans, expansion joints, filter units and plenums. In its response to the NRC staff RAI on TB ventilation system pressure boundary, the licensee states that the exhaust systems take suction from various areas in the TB. While an intact duct is required for effective system operation, it is not essential that the duct pressure boundary be leak tight. Exhaust duct in-leakage does not impact the effectiveness of the exhaust fans in purging the TB. The NRC staff finds the licensee's response to be acceptable.

The EPRI report endorses the SQUG GIP-2 generic bounding spectrum (which bounds the earthquake experience data). Figure 3-1 of LAR Enclosures 11 and 12 shows that the SQUG generic bounding spectrum envelopes the Hatch ½ SME ground motion response spectra over the entire frequency range which in turn envelops the HNP DBE. Therefore, the NRC staff finds the EPRI guidelines are acceptable for the evaluation of the proposed AST TB ventilation systems.

The licensee walked down the AST in-scope TB exhaust ductwork for Units 1 and 2 to screen and identify duct and support seismic vulnerabilities and undesirable conditions that could lead to damage or failure in an earthquake and included the walkdown data sheets, as well as the seismic walkdown evaluation and results in LAR Enclosures 11 and 12. Several duct and duct support outliers were identified as a result of the Unit 1 and Unit 2 walkdowns. The outliers were resolved either by evaluation or by modification. The licensee in its responses to the NRC staff RAI provided outlier evaluation results and acceptability justification. The licensee in its response to the NRC staff RAI also states that all identified modifications have been completed. The NRC staff reviewed the licensee's responses and finds them acceptable.

Three duct runs in Unit 1 were selected for bounding analytical review. These duct runs are representative, worst case bounding samples of different types of ductwork and supports for the HVAC duct systems reviewed during the walkdowns of Unit 1 and Unit 2 and are documented on the HVAC Duct System Analytical Review Data Sheets in Attachment C of LAR Enclosure 11. As stated in Section 2.3.3.1 of this SE, the in-structure response spectra accepted for resolution of USI A-46 for HNP is ½ SME ISRS. Therefore, the licensee used the full SME acceleration values multiplied by ½ for determining seismic accelerations for the analytical review calculations with the following modification. The full SME ISRS are specified at 5 percent (%) damping. The applicable damping for calculating the duct seismic stress is 7% per Section 4.2 of LAR Enclosure 13. Therefore the 5% damping peak spectral acceleration values are multiplied by the square root of the damping ratios per Section 4.4.3 of the SQUG GIP-2 to obtain 7% damping values for analysis of the ductwork. The ductwork was analyzed using the response spectrum method. The NRC staff finds this seismic analysis approach acceptable.

The duct evaluation is based on a normal allowable bending stress of 8 thousand pounds per square inch (ksi) for rectangular galvanized steel duct per Section 4.2.1 of the EPRI guidelines. The duct bending maximum stress for the dead load plus ½ SME seismic load case was less than 8 ksi for all three ductwork sections analyzed and are, therefore, acceptable.

The SRT selected five representative duct supports for analytical review and provided a summary of the results in Section 5.3 of LAR Enclosures 11 and 12 supplemented by its responses to the NRC staff RAI. The analyzed supports met the criteria of Appendix F of the EPRI guidelines and are, therefore, acceptable.

In its response to the NRC staff's RAI, the licensee indicated that there are no air handling units or instrumentation and control (I&C) cabinets which need to be seismically qualified for the proposed AST analysis. The licensee provided a commitment in LAR Enclosure 7 to complete walkdowns of the TB motor control centers (MCCs) associated with crediting the Units 1 and 2 turbine exhaust ventilation by May 31, 2008, to confirm that they are seismically rugged to maintain their functionality under the plant design basis SSE. In response to a NRC staff RAI, the licensee indicated in its letter of July 14, 2008 (Reference 29), that MCC 1R24-S016 and its included distribution panel 1R25-S120 for Unit 1 and MCC 2R24-S016 and its included distribution panel 2R25-S106 for Unit 2 were "evaluated for seismic acceptability per the above commitment." The licensee states that its evaluation of these MCCs was completed "using the appropriate SQUG criteria, and determined to be acceptable for the applicable seismic design requirements "as-is" with no need for modification." SNC provided clarification in its letter of August 14, 2008, that the criteria for the walkdown was revision 2 of the Generic Implementation Procedure. The NRC staff concludes that this resolves this issue, since the evaluations were completed using acceptable criteria and the results were found by the licensee to meet the criteria requirements.

The licensee has employed the EPRI guidelines of LAR Enclosure 13 (supplemented by the peer review comments of LAR Enclosure 14) which is similar to GIP-2 and its associated NRC SE to demonstrate the seismic ruggedness for the non-seismically analyzed HNP TB ventilation systems that are credited in the proposed AST analysis, Therefore, based on its review as summarized above, the NRC staff concurs with the licensee that the in-scope TB ventilation systems of HNP Units 1 and 2, are seismically rugged for the proposed AST.

2.3.4 Summary

Based on the above evaluation, the NRC staff concludes there is reasonable assurance that the HNP Unit 1 main condenser and turbine building, Units 1 and 2 secondary containment bypass leakage piping and supports, Unit 1 MSIV ALT pathway piping and supports and Units 1 and 2 turbine building ventilation exhaust systems, are seismically rugged to maintain their functionality under the Hatch design basis earthquake.

It should be noted that the acceptance of the GIP-2 and earthquake experience data methodology as presented by the licensee in its LAR is applicable only for ensuring the pressure boundary integrity of the alternate leakage treatment path and the secondary leakage bypass path and ensuring the seismic ruggedness of the proposed AST credited HVAC systems, supports and components, and is not an endorsement that the experience-based methodology discussed in GIP-2 is applicable for other applications at HNP other than those approved by the NRC staff in conjunction with the resolution of USI A-46.

2.4 Plant Systems

This portion of the review is based on licensee submittals dated August 29, 2006 (Reference 1), and June 22, 2007 (Reference 5). SNC requested NRC approval of the Main Steam Isolation Valve (MSIV) leakage pathway configuration in conjunction with their request to implement use of an AST at the HNP. This evaluation applies only to the proposed MSIV leakage pathway configuration.

The acceptability of the licensee's amendment request with regard to the MSIV leakage pathway is based on the guidance provided in the approved General Electric Topical Report, NEDC-31858P-A, Revision 2, "BWROG Report for Increasing MSIV Leakage Limits and Elimination of Leakage Control Systems," (Reference 19).

Regarding the MSIV leakage pathway configuration, the licensee has selected the alternate leakage treatment (ALT) pathway in accordance with the criteria discussed in the approved GE topical report. The GE topical report discusses a method for demonstrating seismic ruggedness of non-Class I structures, systems, and components (SSCs) in withstanding the loading of a safe shutdown earthquake (SSE). Further, this topical report describes an acceptable ALT pathway, and anticipates the need for potential manual actions to establish the required configuration. Accordingly, the pathway and the manual actions needed to establish the configuration are acknowledged in the NRC safety evaluation as acceptable, provided that functional reliability is demonstrated for the ALT pathway on a plant-specific basis.

The licensee depicts the proposed MSIV ALT pathway and associated boundary valves on Figure 1 of Reference 5. The manual actions required to establish this configuration are described in LAR Enclosure 8. According to the licensee, certain post-LOCA manual isolation actions are required to minimize the leakage past the ALT pathway boundary. Also, one valve will be opened manually to establish one of the leakage pathways to the main condenser. The licensee has evaluated the necessary manual actions and determined that HNP personnel have sufficient time to perform the manual actions before increased radiation exposure would become a concern. The NRC staff's evaluation of the proposed ALT pathway and its functional reliability are presented in the following subsections.

One-inch Drain Path

Section 4 of Appendix C of the GE topical report states that if any one-inch (or larger) drain line remains open (without an orifice), essentially all of the release will be via the main condenser. The HNP ALT pathway meets this criteria because the preferred flow path is one-inch or larger and remains open without an orifice. Operator action in the main control room will be required to open a valve to initiate the flow path to the condenser. Class 1E power is supplied to this valve, so its operation does not rely solely on the availability of offsite power. A secondary path to the condenser with an orifice exists. Therefore, essentially all of the HNP MSIV leakage release will be via the main condenser. The NRC staff considers this to be acceptable.

Boundary Isolation Valves

Section 5.3 of the SE for the GE topical report requires licensees to provide assurance of the MSIV ALT Pathway boundary valves reliability. Referring to LAR Enclosure 8, the licensee has clearly described the boundary isolation valves and associated manual actions to configure the MSIV ALT pathway. The latest revision of Figure 1 in Reference 5, in concert with the information in LAR Enclosure 8, delineates the scope of the ALT pathway. There is a total of 10 boundary valves that require operator actions to close; four are locally closed, and six are remotely closed to isolate the ALT pathway from the unanalyzed piping. In the event of a LOCA, the MSIVs, the turbine stop valves and the turbine bypass valves will automatically close. The reactor feed pump turbine (RFPT) stop valves will close on an automatic or manual trip of the RFPTs.

The NRC staff reviewed these boundary isolation valves and the associated manual actions and found them to be acceptable as proposed. The actions will assure that MSIV leakage will reach the main condenser via the proposed ALT pathway, which the NRC staff finds acceptable.

Based on the considerations above, the NRC staff concludes that the licensee has established an appropriate MSIV leakage pathway to the main turbine condenser and the operator actions necessary to establish that configuration are consistent with the NRC review criteria stated in the GE topical report (Reference 19). Therefore, the NRC staff finds the proposed ALT pathway, boundary valves, and associated operator actions acceptable with respect to the MSIV leakage pathway configuration.

2.5 Containment Review Considerations

This portion of the review is based on the licensee's submittals dated August 29, 2006, February 27, June 25 and July 2, 2008. The following NRC regulatory criteria are applicable to this portion of the NRC staff's review: 10 CFR 50 Appendix A, General Design Criterion 19, "Control Room," 10 CFR 50.36(d)(2)(ii)(C) on Technical Specifications, 10 CFR 54.37(b) on FSAR updates, 10 CFR 50.55a, "Codes and Standards," 10 CFR 50.65 on maintenance requirements, 10 CFR 50.67, "Accident Source Term," RG 1.183, RG 1.29, Revision 4, "Seismic Design Classification," NUREG-0800, Standard Review Plan (SRP) sections 6.4, "Control Room Habitability System", March 2007, SRP section 9.4.1, "Control Room Area Ventilation System", March 2007, SRP section 9.4.4, "Turbine Area Ventilation System", March 2007, and SRP Section 6.2.3, Branch Technical Position Containment Systems Branch 6-3, "Determination of Bypass Leakage Paths in Dual Containment Plants," July 1981.

The licensee's submittal of August 29, 2006, indicated that the licensee would modify the logic for the cable spreading room fan control to automatically trip the supply and exhaust fan on initiation of the pressurization mode in the main control room. Since the licensee had not subsequently indicated the status of that modification, the NRC staff's letter of July 9, 2008, requested the licensee to propose a condition to the license to ensure the completion of that modification. The licensee's letter of July 14, 2008, states that the modification was completed in October 2007, and that the FSAR and applicable drawings have been updated accordingly. Therefore, the NRC staff concludes that a license condition for this item is not required.

2.5.1 Proposed Changes to TS

TS 3.6.1.3 Primary Containment Isolation Valves

The proposed license amendment would also revise the following TS that are associated with the analyses performed to support the AST. The proposed change for Unit 1, adds a new Surveillance Requirement (SR) 3.6.1.3.13, which establishes a maximum combined leakage rate for all secondary containment bypass leakage paths of 0.02La. The proposed change for Unit 2, revises SR 3.6.1.3.10 to increase the maximum combined leakage rate for all secondary containment bypass leakage paths from 0.009La to 0.02La. La is defined in 10 CFR Part 50, Appendix J.

The secondary containment bypass leakage rate assumptions in the radiological dose consequences analysis for the LOCA form the basis for the revised TS limits. The increase in bypass leakage is necessary to allow for newly identified bypass leakage paths. The addition of this TS SR to Unit 1 reflects a required RG 1.183 assumption in the accident analyses and standardizes the TS between units.

The NRC staff's assessment found these changes acceptable since the proposed secondary bypass leakage rate limit of 0.02La was assumed in the accident analysis and regulatory criteria have been met.

Another proposed change is to eliminate the per line MSIV leakage rate limits from the TS SR for both units (SR 3.6.1.3.10 and SR 3.6.1.3.11, respectively). Specifically for Unit 1, the licensee proposes to establish a combined maximum leakage rate of 100 standard cubic feet per hour (scfh) when tested at $\geq$ 28.0 pounds per square inch guage (psig) and < 50.8 psig and for Unit 2, a combined maximum leakage rate is reduced from 250 scfh to 100 scfh when tested at $\geq$ 28.8 psig and < 47.3 psig. The licensee indicates that the pressure values of 50.8 psig and 47.3 psig represent calculated peak drywell pressures for Unit 1 and Unit 2, respectively, in the event of a LOCA.

The revised proposed values for MSIV combined maximum leakage rates are used in the radiological dose consequences analysis for the LOCA. The contribution to total combined leakage from any individual MSIV is not considered in the analysis. The analysis assumes that the maximum allowed combined leakage rate is entirely through one MSIV. The NRC staff found this change acceptable since this value was assumed in the revised accident analysis, and calculated doses are below the regulatory criteria of 10 CFR 50.67.

A second test pressure range, with a corresponding leakage rate criterion, is proposed for both units when test pressure exceeds the peak calculated drywell pressure during a LOCA. This is in addition to the 100 scfh combined maximum leakage rate specification when tested within the specified test pressure range that is below the calculated peak drywell pressure. For Unit 1, a combined maximum leakage rate of 144 scfh is established when tested at $\geq$ 50.8 psig. For Unit 2, a combined maximum leakage rate of 144 scfh is established when tested at $\geq$ 47.3 psig.

The addition of a second MSIV leakage rate criterion for testing at or above calculated peak drywell pressure provides a more accurate leakage rate acceptance criterion for test pressures that are higher than calculated post-LOCA peak drywell pressures. This facilitates testing the MSIVs in the accident direction at peak accident drywell pressure as preferred by 10 CFR 50 Appendix J, as opposed to testing the MSIVs in the reverse direction at a lower test pressure as allowed by existing Hatch Appendix J exemptions. A higher pressure would result in a higher mass flow rate through a given leakage area. The higher leakage rate (mass flow rate) acceptance criterion is based on a pressure and mass flow rate analysis. This allows for the use of a different MSIV Appendix J test configuration as dictated by plant configuration during the outage, while also ensuring that the appropriate acceptance criterion exists for the actual test pressure used.

For Unit 2, the requirement to restore MSIV leakage to 11.5 scfh upon discovery of leakage not meeting the 100 scfh leakage rate limit has been proposed to be eliminated. The NRC staff finds this change to be acceptable since it is not an input or assumption in the radiological dose consequence analysis.

TS 3.7.9 Turbine Building Ventilation (TB HVAC) Exhaust System Fans

The proposed change will add a new TS Limiting Condition for Operation (LCO) 3.7.9 for the TB HVAC exhaust system fans on Hatch 1 and 2, reflecting the crediting of the turbine building purge function as part of the AST assumptions for three of the four Hatch DBAs. One of the four available TB HVAC exhaust system fans is sufficient to deliver the credited purge flow. To account for potential single failures, proposed LCO 3.7.9 will require one Unit 1 and one Unit 2 TB HVAC exhaust system fan to be operable in modes 1, 2 and 3.

SR 3.7.9.1 and SR 3.7.9.2 have been proposed in association with the proposed new LCO 3.7.9. SR 3.7.9.1 requires the operation of each TB HVAC exhaust system fan for greater than or equal to 15 minutes, every 92 days. SR 3.7.9.2 verifies the manual transfer capability to alternate power supply for each TB HVAC exhaust system fan, every 24 months.

The licensee indicated the new associated conditions, required actions, completions times, and surveillance requirements are closely modeled after similar ventilation system TS in the Standard Technical Specifications. The NRC staff assessment found the proposed new TS acceptable in order to ensure compliance with 10 CFR 50.36(d)(2)(ii)(C).

2.5.2 Standby Liquid Control System

The standby liquid control (SLC) system is credited for the injection of sufficient sodium pentaborate (SPB) solution to prevent the re-evolution of iodine from the suppression pool for a 30-day period following a DBA LOCA.

The NRC staff review of the quantity of SPB available with respect to the quantity of acid producing debris and radiolytic acid production to confirm adequate pH control is provided in Section 2.7 of this SE. The NRC staff's review here discusses the SLC system with respect to its role in delivery of sodium pentaborate to the suppression pool for pH control. The control of pH in the suppression pool is required to mitigate the consequences of a DBA in which fuel is damaged. As such, the new role being assigned to the SLC is a safety related role. The licensee states that the SLC is designated as a special safety system or safe shutdown system, and not an ESF system. Therefore, the NRC guideline, "Guidance on the Assessment of a BWR SLC System for pH Control" was used to evaluate the SLC system for its ability to perform its AST function of post-LOCA suppression pool pH control.

The licensee states in their February 27, 2008, submittal that the SLC system meets the following criteria. The following is extracted from the February 27, 2008 submittal:

(1) The SLC system process equipment, instrumentation, and controls essential for post-LOCA injection of SPB solution in the reactor is designed as Seismic Category in accordance with Appendix A to 10 CFR 100 and Regulatory Guide 1.29 (August 1973).

(2) The SLC system is required to be operable in the event of an offsite power failure. Therefore, the pumps, valves and controls are powered from standby AC power supply. The pumps and valves are powered and controlled from separate buses and circuits so that a single failure does not prevent system operation.

(3) The applicable components of the SLC system are inspected and tested in accordance with the American Society of Mechanical Engineers (ASME) Inservice Inspection and Inservice Testing Programs, (ISI and IST) as required by 10 CFR 50.55a, Codes and Standards.

(4) The functions of the SLC system are evaluated in the Hatch Maintenance Rule program consistent with 10 CFR 50.65 to provide reasonable assurance that the system will perform reliably.

(5) The post-LOCA environment in which the SLC system would be required to operate has been determined to be a mild environment. Cables associated with the SLC system were evaluated and determined to be environmentally qualified for the SLC post-LOCA mission. Therefore, the SLC system meets the requirements of 10 CFR 50.49 and Appendix A to 10 CFR Part 50.

Based on its review of the information above, the NRC staff concludes that the SLC system is appropriately designated a safety related system, and is designed and installed as a high quality system that provides reasonable assurance that the sodium pentaborate will be injected into the core upon activation.

The licensee indicated that applicable plant procedures will be revised and implemented during the AST implementation phase, as necessary so that upon detection of high drywell radiation associated with the postulated activity release, manual initiation of SLC injection is executed for a LOCA to maintain suppression pool pH at or above 7.0. The impacted procedures include the severe accident guidelines (SAGs), abnormal operating procedures, system operating procedures, and annunciator response procedures. The NRC staff's review of the proposed changes to plant procedures that implement SLC injection of SPB and any plant personnel training required is provided in section 2.6.

The NRC staff considered components that could be subject to single failure. The licensee identified two potential active failures that could impact the SLC system. The first potential failure is the SLC initiation control switch located on the Unit 1 and Unit 2 panels 1H11-P603 and 2H11-P603, respectively, in the main control room. The second potential failure is one of the two check valves in series on the injection line that are credited to change state to inject the SPB solution.

The licensee states that the injection line check valves are stainless steel Rockwell Edward 1½-inch Piston Lift Check Valves, mounted horizontally in the injection line. The injection line check valves were procured as Class 1 and were designed to open against full reactor pressure. Leak rate testing to verify the integrity of the injection check valves is performed once per operating cycle per the Inservice Inspection Program. A review of the Hatch maintenance history for the SLC system did not discover any failures of the check valves to open or close on demand. A review of industry databases, Equipment Performance and Information Exchange (EPIX) and Nuclear Plant Reliability Data System (NPRDS), was performed by the licensee and no failures of check valves of this manufacturer and type failing to open was identified. Although acknowledging that a single failure to open of one of the two check valves could prevent SLC injection, the NRC staff has determined that the potential for failure is very low based on the quality as established by its procurement, periodic testing and inspection, and historical performance of the component. The NRC staff finds that the use of a single failure penetration of the containment with the identified check valves as described by the licensee to be acceptable.

The licensee states that the SLC system is actuated by a key-locked, three position switch. The SLC initiation control switch was procured as Class 1E and is a General Electric (GE) Type SB-1 five-stage rotary cam-operated switch. The five individual stages are nested into each other on a common operating shaft with a common fixed contact support and front and rear support. The stack is tied together with two tie bolts threaded into the front support. Each stage has two contacts. The entire assembly is enclosed in a metal cover that provides protection for the contacts. SLC system functional testing is performed once per operating cycle, to verify flow through one SLC subsystem from pump to reactor pressure vessel. The functional test is on a staggered test basis and alternates each subsystem tested. During the functional test, operation of the control circuits, indicators, and the alarm annunciator are verified. The licensee's search of the industry databases, EPIX and NPRDS, for GE SB-1 switches found 24 failures of various modes for that type switch. There were failures involving dirty or corroded contacts, loose parts, sticking and binding, interferences holding contacts open and normal age wear. Three of the 24 failures found were at Hatch. These failures were attributed to (1) dirty contacts, (2) contacts out of adjustment, and (3) a loose cam shaft, however, none of these failures were with the SLC initiation switch. The SLC initiation control switch is located in the main control room, an environmentally controlled facility, which should reduce or eliminate the occurrence of contact corrosion. The NRC staff acknowledges that the selector switch in the main control room could fail and prevent the initiation of either SLC sub-system. The NRC staff has determined that the switch is a highly reliable component at an accessible location. In the unlikely event of a failure of the control switch, a repair of the switch could be attempted in the main control room, considering that SLC injection is not required for the first 2 hours, and an additional compensating action is the ability to install jumpers to overcome failure of the control switch.

The NRC staff considered the transport of the sodium pentaborate from the reactor vessel to the suppression pool. The SLC system injects the sodium pentaborate to the reactor vessel. The transport of reactor vessel contents including the sodium pentaborate to the pool is by flow through the break (assumed to be a recirculation line break) to the drains that feed the suppression pool.

The NRC staff concluded that there would be mixing and transport at some rate and that it was reasonable to assume the concentration of sodium pentaborate in the core would equalize with the concentration in the suppression pool within an acceptable time after SLC injection. As a consequence, there would be sufficient pH control to deter and prevent iodine re-evolution.

2.5.3 Turbine Building Ventilation System

The turbine building (TB) ventilation system is credited with purging the area around the main control room beginning 9 hours following a LOCA, CRDA, and MSLB for dose mitigation. The TB ventilation system is currently categorized as a non safety related system, but the licensee is proposing a safety related function in its AST analysis. The NRC staff conducted a review of the TB ventilation system to assess the adequacy of the system to meet the new safety related function. The following areas of review and information were considered by the NRC staff.

The NRC staff's review of the seismic and structural ruggedness of the TB ventilation system with respect to the system being capable of performing its function of dose mitigation considering a seismic event is provided in section 2.3. The NRC staff's review of the capability of powering the TB ventilation system by an emergency power source with respect to the system being capable of performing its function of dose mitigation considering a LOCA concurrent with a loss of offsite power (LOOP) is provided in section 2.2. The NRC staff review of the planned changes to plant procedures that implement the dose mitigation function for the TB ventilation system and associated training for plant personnel is provided in Section 2.6.

The NRC staff review of the TB ventilation system with respect to its role in purging the area around the main control room by TB exhaust ventilation for dose mitigation is discussed here. The mitigation of dose around the control room is required to mitigate the consequences of a LOCA, CRDA, or MSLB. As such, the new role being assigned to the TB ventilation system is a safety related role.

The NRC staff reviewed the licensee's submittals dated August 29, 2006, February 27 and June 25, 2008, on the use of the TB ventilation system for the safety related function. From the licensee statements, the NRC staff concluded the following:

The function of inspection and testing is to provide periodic continued assurance that a system continues to be able to perform its credited safety function by identifying any system degradation due to various wear mechanisms through the testing/inspection process. The licensee has evaluated the monitoring warranted for the TB HVAC exhaust systems to demonstrate that it continues to be able to perform its credited purge function. The licensee intends to add the TB HVAC exhaust systems to the scope of the Hatch Maintenance Rule program, in accordance with 10 CFR 50.65, and to the scope of license renewal, in accordance with 10 CFR 54.37(b). Inclusion in the scope of license renewal will result in periodic monitoring of the condition of the TB HVAC exhaust systems ductwork supports.

Beyond inclusion in the scope of the Maintenance Rule program and license renewal, the licensee's evaluation indicates that minimal testing/inspection is warranted based on the existence of minimal wear mechanisms and the TB HVAC exhaust systems' required continuous operation during normal plant operation. Unlike a system with water or steam as a process fluid, TB HVAC exhaust systems ductwork is subject to minimal wear. The licensee states that the adequacy of fan flow can be monitored from the main control room and is demonstrated by daily performance of TB HVAC exhaust systems design functions, such as removal of heat dissipated from plant equipment and air movement and filtering to control potential airborne radioactivity, by two of the available four fans on the two Hatch units. Since the TB HVAC filter units are not credited in the AST analyses, no filter testing is necessary to demonstrate filtration capabilities in support of AST DBA analyses.

In addition, in a letter dated June 25, 2008, the licensee proposed to add TB ventilation exhaust system fans, "LCO 3.7.9," to the TS, in order to ensure compliance with CFR 50.36(d)(2)(ii)(C), since the TB purge function is credited to mitigate three of the four Hatch DBAs. To account for potential single failures, proposed LCO 3.7.9 will require one Unit 1 and one Unit 2 TB HVAC exhaust system fan to be operable in TS Modes 1, 2 and 3. Applicable required actions, completion times, and surveillance requirements were also proposed in association with LCO 3.7.9. The NRC staff finds this proposed new TS LCO to be acceptable as stated in Section 2.5.1 of this safety evaluation.

The NRC staff reviewed the scope of HVAC ductwork and components that must survive seismic affects to deliver the flowrate required for the licensee's accident analysis assumptions. The licensee defined the scope of work, sufficient to perform the safety function as the normal TB ventilation return air ductwork from intake registers in the TB to the main exhaust plenum in the reactor building. The extent of the scope of work includes ductwork, duct supports and associated in-line components such as registers, dampers, damper actuators, in-line fans, expansion joints, filter units and plenums. The NRC staff reviewed the TB ventilation drawings submitted by the licensee and finds the licensee's defined scope of work to be acceptable.

The NRC staff considered components that could be subject to single failure. The licensee indicated that no single failures exist that would impact the TB exhaust capability of both units. The only failure mechanism that could affect both units is a seismic event. The air piping system supplies both safety and non-safety/non-seismic systems. A failure of the air systems of both Unit 1 and Unit 2 would render both TB ventilation systems incapable of performing their required exhaust functions. The licensee states that a modification will be completed to ensure that a loss of air event does not render both TB exhaust systems incapable of operating. Specifically, the air supply for both units TB HVAC exhaust system dampers will be changed from interruptible instrument air to non-interruptible instrument air. The licensee states that this modification will be implemented by December 31, 2009. This issue has also been included as a condition to the license.

The NRC staff considered purging the area around the main control room. For each unit, air is exhausted from the TB by a duct system to the outside environment via the reactor building vent plenum by two exhaust fans. The exhaust from the TB is filtered by two 50% capacity filter trains. Each filter train is rated for 15,000 cubic feet per minute (ft^3/min) and consists of a bank of prefilters, carbon adsorbers, and high efficiency particulate air filters. The carbon adsorber bank is provided with a water deluge system. Only one of the two 100% capacity exhaust fans is normally operating. If the operating exhaust fan fails, the standby exhaust fan starts automatically and an alarm is annunciated in the main control room. In the event of a LOCA, CRDA, or MSLB, the appropriate operating procedures will be changed to ensure that TB exhaust ventilation (one of the four TB exhaust fans) is initiated within 9 hours of the start of the accident, in accordance with the design basis assumptions for TB ventilation used in the DBA analyses.

The NRC staff concluded that there would be purging of the area around the main control room and that it is reasonable to assume that airborne radioactivity would be minimized at some time following the DBA.

Based on a review of the licensee's submittal and the licensee's responses to the NRC staff's RAIs, the NRC staff finds that the TB ventilation system as proposed for the new dose mitigation function is acceptable because there is reasonable assurance that the system would be available to mitigate dose as defined in the licensing basis for DBAs.

2.5.4 Secondary Containment Bypass Leakage

The primary containment leakage that bypasses the secondary containment (reactor building) is assumed to be into the condenser for evaluating MCR doses for the DBA LOCA analysis. Activity holdup and deposition in the condenser from this secondary containment bypass leakage is credited in a manner similar to the treatment of MSIV releases and the MSIV alternate leakage treatment pathway.

The licensee performed an evaluation of the Unit 1 secondary containment system using the guidance provided by Branch Technical Position CSB 6-3, "Determination of Bypass Leakage Paths in Dual Containment Plants." All primary containment penetrations were assessed to identify the leakage paths that do not terminate within the secondary containment and should be considered as potential secondary containment bypass leakage paths. Table 2 to Enclosure 1 of the licensee's submittal lists the piping systems identified as potential bypass leakage paths.

The evaluation of bypass leakage identifies bypass leakage paths and determines leakage rates. Potential bypass leakage paths are formed by penetrations which pass through both the primary and secondary containment boundaries and which may include a number of barriers to leakage (e.g., isolation valves, seats, gaskets, and welded joints). While these barriers reduce they do not necessarily eliminate leakage. Therefore, in identifying potential leakage paths each of these penetrations should be considered together with the capability to test them for leakage in a manner similar to that of the containment leakage tests required by 10 CFR Part 50, Appendix J.

The NRC staff finds the scope of potential secondary containment bypass leakage paths identified by the licensee acceptable, based on the licensee using the guidance of Branch Technical Position CSB 6-3, which is also consistent with the current licensing basis.

2.5.5 Use of Potassium Iodide (KI) for an Interim Period

The Hatch current licensing basis main control room unfiltered inleakage limit is 110 cfm based on having a pressurized control room, with the administration of KI tablets to main control room occupants within 2 hours after the start of a design basis LOCA.

By letters dated August 4, 2003, March 29, 2004, October 27, 2004, and November 10, 2005, the licensee submitted a course of action for developing responses to NRC Generic Letter (GL) 2003-01, "Control Room Habitability" information requests for Hatch. GL 2003-01 was written to inform licensees that the design basis assumptions used for control room unfiltered inleakage, even with a pressurized control room, could be non-conservative. This was validated through testing at several power reactor facilities using the standard test method described in American Society for Testing and Materials (ASTM) consensus standard E741, "Standard Test Method for Determining Air Change in a Single Zone by Means of a Tracer Gas Dilution."

In order to address the possibility of unfiltered inleakage into the Hatch control room, the incorporation of KI was approved by the staff, by letter dated May 25, 2006, on an interim basis as a measure to limit the thyroid dose to control room occupants in the event of a design basis LOCA. The incorporation of KI in the interim licensing basis is provided to assure that the 30-day thyroid dose remains within the regulatory limits of 10 CFR 50, Appendix A, General Design Criterion (GDC) 19, with main control room unfiltered inleakage up to 110 cfm. As a condition of the current licensing basis, KI was credited in the amendment issued on May 25, 2006, to limit post-LOCA doses to main control room personnel for an interim period, expiring on May 31, 2010.

By letter dated July 2, 2008, the licensee proposed to extend the period for incorporating the use of KI into the Hatch licensing basis from May 31, 2010, to May 31, 2012 for HNP Unit 1 and from May 31, 2010, to May 31, 2011, for HNP Unit 2. Specifically, the date of May 31, 2010, in existing license conditions 2.C(8) for HNP Unit 1 and 2.C(3)(f) for HNP Unit 2, that was authorized by the NRC in the license amendment issued on May 25, 2006, is proposed to be extended to continue to authorize the licensee to credit administering KI to reduce the 30 day post-accident thyroid radiological dose to the operators in the main control room until May 31, 2012, for HNP Unit 1 and until May 31, 2011, for HNP Unit 2. The licensee also proposed to modify the license conditions by deletion of the words "rendering the crediting of potassium iodide no longer necessary" and "update" from the license conditions. SNC withdrew those changes to the license condition text in its letter of August 14, 2008. The NRC staff finds the withdrawal of those text changes acceptable.

The licensee initially proposed that the interim period be defined as approximately 4 years, with the license condition including a specific end date for crediting KI of May 31, 2010. This period of time was intended to allow for 1) tracer gas testing of the Hatch 1 & 2 common control room, 2) submittal of AST for NRC review, 3) NRC review of the licensee AST submittal in a projected time period of 2 years or less, and 4) if needed, additional time to complete the NRC review or for the licensee to initiate additional design basis changes, including potential plant modifications that may be identified as necessary. While 1, 2, and 3 have been accomplished, as a result of the NRC AST review, new modifications have recently been identified and the licensee has concluded that the recently identified modifications for AST can not be completed by May 31, 2010.

In a letter dated February 25, 2008, the licensee identified that modifications would be required to certain alternative leakage treatment (ALT) path boundary valves in order to provide the capability to remotely position the valves to their required post-accident position. The letter identified that the proposed modifications could not be implemented until May 31, 2012, for HNP Unit 1 and until May 31, 2011, for HNP Unit 2. The extension of time that the licensee has proposed to credit administering KI to reduce the 30 day post-accident thyroid radiological dose to the operators in the main control room is tied to the boundary valve modification implementation dates.

The licensee states that the radiological consequences analyses remain as described in the enclosure to the licensee's March 17, 2006 submittal to incorporate license conditions providing for crediting of KI for the original interim period through May 31, 2010. As indicated in its letter dated May 25, 2006, the NRC staff has previously reviewed the regulatory and technical analysis related to the radiological consequences of a DBA LOCA including the assumptions, inputs, and methods used by the licensee to assess the impact of crediting KI. In addition, the NRC staff performed independent calculations to confirm the conservatism of the licensee's analyses. It was the staff's finding, with reasonable assurance, that the licensee's estimates of the control room thyroid doses, when crediting dose reduction from KI, will continue to comply with regulatory requirements and guidance documents.

The NRC staff finds the proposed revisions to license conditions that authorize the crediting of KI for an extended period not to exceed May 31, 2012, for HNP Unit 1 and May 31, 2011, for HNP Unit 2, to be acceptable. This condition is consistent with regulatory guidance that states that the prophylactic use of KI can be credited as an interim compensatory measure while final corrective actions are being taken.

2.6 Human Factors

SNC's license amendment request (LAR), proposed to credit four new operator actions in HNP's safety analysis. The NRC staff requested additional information in its letter dated December 28, 2007, and SNC responded in its letter dated February 25, 2008 (reference 13).

2.6.1 Regulatory Evaluation – Human Factors

The NRC staff reviewed the licensee's overall request using the guidance contained in RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors;" and NUREG-0800, "Standard Review Plan," Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms." With regard to the proposed new operator actions, the NRC staff used the guidance contained in NRC Information Notice (IN) 97-78, "Crediting Operator Actions in Place of Automatic Actions and Modifications of Operator Actions, Including Response Times;" RG 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Environs Conditions During and Following An Accident," Rev. 2; NUREG-0800, "Standard Review Plan," Chapter 18.0, "Human Factors

Engineering;" NUREG-0737, “Clarification of TMI Action Plan Requirements;” and the HNP’s current licensing basis, as contained in the HNP Final Safety Analysis Report (FSAR).

2.6.2 Technical Evaluation – Human Factors

Proposed New Operator Actions for Credit

To support the request to implement an AST at HNP, SNC performed a re-analysis of selected design-basis accidents, consistent with the requirements of 10 CFR 50.67, "Accident Source Term," and NRC guidance documents (e.g., RG 1.183). Contained within this re-analysis were four assumptions regarding operator actions which differed from the current licensing basis:

1. Initiate the standby liquid control system (SLCS) within 2 hours following a design basis loss of coolant accident (DBLOCA). This action was assumed in order to maintain the primary containment suppression pool pH at or above 7, such that the re-evolution of iodine from the suppression pool would be prevented.

2. Initiate drywell spray within 15 minutes following a DBLOCA. This action was assumed to help remove airborne particulates in the primary containment drywell, and for primary containment temperature and pressure reduction.

3. Place the main steam isolation valve (MSIV) alternate leakage treatment (ALT) flow pathway in-service within 90 minutes following a DBLOCA on Unit 1. MSIV ALT has been approved for Unit 2 in reference 25. This action was assumed to allow credit for a reduction in the release past the MSIVs, due to holdup and retention in the main steam line piping downstream of the MSIVs and in the main condenser.

4. Start turbine building exhaust ventilation within 9 hours following three different accidents: DBLOCA, CRDA , and a main steam line break (MSLB). This action was assumed to allow for removal of activity from the turbine building, which has a direct leakage path to the main control room (MCR).

SNC has requested that the NRC approve these four new operator actions, such that these four new actions become a part of the revised licensing basis for HNP.

NRC Bases for Approval for the New Operator Actions

Using the regulatory guidance mentioned in Section 2.6.1 above, the NRC staff determined that the four new operator actions were acceptable for credit in HNP’s safety analysis with regards to human factors, based upon:

1. The new operator actions are prompted by appropriately qualified indications and alarms in the MCR.

All of the operator actions associated with a DBLOCA (initiate SLCS and drywell spray, place MSIV ALT in-service, start turbine building exhaust ventilation) will be prompted by operator identification of a DBLOCA, which is already credited at HNP, accompanied by a MCR alarm and MCR indications of high radiation levels in the primary containment drywell. Drywell radiation levels will be measured on each Unit's post-accident (wide range) primary containment radiation monitoring system, which consists of two drywell radiation sensors per Unit, associated circuitry, a recorder located in the MCR for each sensor, and input into a MCR alarm. All components receive electrical power from station emergency power sources and will be available should a loss of offsite power (LOSP) occur. In addition, the post-accident (wide range) primary containment radiation monitors are fully qualified in accordance with RG 1.97 as Category 1, and are required to be operable per HNP technical specification (TS) 3.3.3.3.1.

The operator action to start a turbine building exhaust ventilation fan for the other two DBAs, CRDA and MSLB, will be prompted by the primary indications and alarms used to identify the accident (e.g., reactor power excursion followed by a reactor scram, rod position indication; high main steam line flow, low reactor pressure, reactor scram, MSIV closure) accompanied by a MCR alarm and MCR indications of high radiation in the turbine building. Each of these instruments, including the turbine building area radiation monitors, contain multiple redundant channels, are available following a LOSP, and are appropriately qualified in accordance with RG 1.97.

2. <u>The new operator actions to be credited are uncomplicated and can be completed in the assumed times.</u>

Initiate SLCS within 2 hours following a DBLOCA

To initiate SLCS on a given Unit, a single key-locked hand switch is operated from the MCR. To address the potential for a single active failure of the key-locked hand switch, SNC states that pre-staged electrical jumpers would be installed to bypass the hand switch. For a given Unit, the jumpers would be installed on the back of the MCR panel where the SLCS hand switch is located, which is readily accessible from the MCR (i.e., no panel doors or covers need to be opened or removed). Based upon a preliminary review, the licensee has determined that from two to five jumpers will be required to jumper out a Unit's SLCS hand switch, depending upon whether multi-point or single-point jumpers are utilized.

Operating a single hand switch can easily be completed within 2 hours, and the licensee has estimated that installing jumpers, if required, would also be a short time when compared to the allowed 2 hours.

Initiate drywell spray within 15 minutes following a DBLOCA

The current licensing basis at HNP credits one residual heat removal (RHR) pump operating in suppression pool cooling at 10 minutes following a DBLOCA. The licensee proposes to credit this same RHR pump for drywell spray, which will require operator action to alter the returning RHR flow path from the suppression pool to the drywell spray header. The operator actions, all performed from the MCR, to accomplish this, as described in reference 13, are:

(1) Place the Containment Spray Valve Control 2/3 Core Height Permission Switch in the Manual Override position (if required due to low reactor water level)

(2) Place the Containment Spray Valve Control switch in the Manual position.

(3) Confirm the operating RHR pump is running.

(4) Close or throttle the RHR to Suppression Pool Return Valve.

(5) Open the Containment Spray Inboard Valve.

(6) Throttle open the Containment Spray Outboard Valve to establish flow.

Shifting one RHR pump from suppression pool cooling to drywell spray requires only a few switch manipulations, and according to the licensee, can be readily accomplished within the credited overall 15 minute response time.

Place MSIV ALT in-service within 90 minutes following a DBLOCA on Unit 1

Assuming a concurrent LOSP with a DBLOCA, to place MSIV ALT in-service on Unit 1 will require operator action to open/close five valves from the MCR, and close four MSIV ALT boundary valves locally:

(1) Open main steam line drain valve 1B21-F021 to establish the flow path to the main condenser. This is a motor operated valve, supplied with Class 1E electrical power (available during a LOSP), and operated from the MCR.

(2) Close steam-to-moisture separator reheater valves 1N38-F101A and 1N38-F101B. These are motor operated valves, and will be modified by May 31, 2012, so that they can be closed from the MCR during a LOSP.

(3) Close steam jet air ejector steam equalizing/drain valves 1N11-F039 and 1N11-F041. These are manual valves which must be closed locally.

(4) Close steam seal feed valves 1N33-F012 and 1N33-F013. These are motor operated valves, and will be modified by May 31, 2012, so that they can be closed from the MCR during a LOSP.

(5) Close steam drain header isolation valves 1N11-F043 and 1N11F044. These are manual valves which must be closed locally.

The operation of the MCR valves can be accomplished within a few minutes after accident identification, and based upon in-plant walk downs at HNP, the licensee has established 24 minutes as the time needed to locally close the four MSIV ALT boundary valves. Taking into account 15 minutes for accident identification prior to any operator actions, and allowing time for the local operators to don protective clothing and be briefed prior to the local actions, it is reasonable to expect that MSIV ALT can be placed in-service within the allowed 90 minute response time.

MSIV ALT has been approved for Unit 2 in reference 25. The licensee has identified that a group of 4 valves (2N11-F004A, 2N11-F004B, 2N33-F003 and 2N33-F004) will be modified on HNP Unit 2 for reasons similar to those discussed above.

Start TB exhaust ventilation within 9 hours following a DBLOCA, CRDA or a MSLB

In SNC's LAR, both units' TB exhaust ventilation systems relied upon the availability of both offsite electrical power and interruptible service instrument air. However, within SNC's RAI responses dated October 18, 2007, and February 25, 2008, SNC proposed to: (1) install a modification to allow for both Units' TB ventilation exhaust fans to be powered from emergency power sources, such that electrical power to the fans would be available should a LOSP occur, and (2) install a modification such that TB ventilation exhaust dampers are supplied operating air from non-interruptible instrument air.

With these two modifications in-place, for the accidents which require starting TB exhaust ventilation (DBLOCA, CRDA, and MSLB) and assuming a concurrent LOSP, the required operator actions are:

(1) Align one TB ventilation exhaust fan from either Unit to a station emergency power source, via one or two manual transfer switches at the fan's motor control center (MCC). The MCCs for the TB ventilation exhaust fans are located in each Unit's reactor building. In accordance with SNC's RAI response, dated October 18, 2007, the design for the transfer of electrical power to the TB ventilation exhaust fans has not been finalized, and there are currently differences between the Units regarding how motive power and control power are supplied to the fans. As such, the number of manual electrical transfer switches to be operated and whether the fans will be started locally or from the MCR has not been firmly established. However, in either case (i.e., operate one additional or less transfer switch, start a fan from the MCR or locally) the operator actions are straightforward and appropriate for credit in HNP's safety analysis.

(2) Start the selected TB ventilation exhaust fan, either locally at the MCC or from the MCR. With available operating air, the exhaust dampers will automatically assume the proper positions.

The assumed operator action time of 9 hours allows for ample time to identify the accident, and also allows ample time for any local and/or MCR actions to start a TB ventilation exhaust fan.

3. <u>Environmental conditions are acceptable for crediting the new operator actions</u>

In accordance with HNP's current licensing basis and their LAR, the MCR will maintain an acceptable human environment (i.e., temperature, humidity, and radiation dose) to allow crediting the MCR operator actions, which include initiating SLCS, initiating drywell spray, MSIV ALT valve manipulations, and potentially starting a TB ventilation exhaust fan.

With regard to local operations, which include closing four MSIV ALT boundary valves and aligning electrical power to and potentially starting a TB ventilation exhaust fan, the licensee determined that the environmental conditions will be acceptable to allow crediting the local operator actions. In particular, the licensee determined that the local operator actions can be completed with operator exposures of 5 rem total effective dose equivalent (TEDE) or less, thus satisfying the requirements of 10 CFR 50, Appendix A, General Design Criterion 19, and NUREG-0737, Item II.B.2.

4. Verifying the success of the new operator actions is straightforward

In addition to indications which prompt operator actions, NRC IN 97-78, “Crediting of Operator Actions in Place of Automatic Actions and Modifications of Operator Actions, Including Response Times,” also states that indications should be available to determine the success of those actions. Even assuming a concurrent LOSP, the following indications will be available:

SLCS success	Multiple indications are available in the MCR, including system pressure, storage tank level, pump running lights, and squib valve continuity lights.
Drywell spray success	Multiple indications are available in the MCR, including RHR pump running lights, RHR valve position indicating lights, drywell spray flow, and the effect of the spray on drywell pressure and temperature.
MSIV ALT success	Success will be indicated by establishing the correct valve line-up. Valve position indicators are available at the locations where the valves are manipulated (MCR or locally).
TB exhaust ventilation success:	For Unit 1, indications are available in the MCR for confirming exhaust fan damper positions and the clearing of the turbine building exhaust fan low flow alarm. For Unit 2, a flow recorder in the MCR gives indication that exhaust air is flowing.

5. The overall integrated response appears to be within the capabilities of an operating crew

In NRC’s RAI dated December 28, 2007, the licensee was asked to evaluate the ability of an operating crew to complete all of the new actions in an integrated fashion. In their response dated February 25, 2008, SNC states, in part, that: (1) initiation of drywell sprays and SLCS require very little time to complete, even if installing SLCS hand switch jumpers are required, and all actions occur in the MCR; (2) placing MSIV ALT in-service requires the local operation of four boundary valves, and on-shift operators with health physics support can complete this task within the specified 90 minutes; (3) the extended time of 9 hours to start TB exhaust ventilation gives an operating crew a high degree of flexibility as to when to take this action, and even with a LOSP, the transfer of electrical power to an exhaust fan is straightforward; and (4) except for the initiation of drywell spray, all other actions are required to be performed after the Emergency Response Organization is required to be staffed.

6. The operator actions will be placed in the appropriate plant procedures

In their RAI response dated February 25, 2008, SNC states that as part of the AST implementation phase, plant procedures will be reviewed and changes made as necessary to direct the new opera/tor actions associated with AST. Specifically for a DBLOCA (wherein drywell radiation occurs (200,000 R/hour)), plant procedures will be revised as necessary to direct operators to: (1) initiate SLCS and inject the entire contents of the SLCS tank, (2) initiate drywell sprays and leave sprays in operation to allow for adequate containment mixing, (3) place MSIV ALT in-service, and (4) start TB exhaust ventilation, including local actions to align the alternate electrical power source to an exhaust fan, if required. With regard to a CRDA and MSLB, plant procedures will be revised as necessary to direct operators to start TB exhaust ventilation if high TB radiation occurs.

7. <u>Operators will receive training on the operator actions</u>

In their RAI response dated February 25, 2008, SNC states that as part of the AST implementation phase, operators will receive training on the new operator actions associated with AST, including the basis for any new system functions (e.g., SLCS credited for suppression pool pH control, drywell spray credited for removal of airborne particulates).

2.6.3 <u>License Conditions and Regulatory Commitments</u>

In support of this section of the SE, license conditions 2.C(9)(1), 2.C(9)(2), 2.C(9)(3), and 2.C(9)(4) are added to HNP Facility Operating License DPR-57 and similarly numbered conditions for HNP Facility Operating License NPF-5, as discussed in section 3.0 below.

<u>Proposed Regulatory Commitments</u>

In addition to the licensee conditions discussed above, the licensee has made the following regulatory commitments:

1. As described in SNC's RAI response dated February 25, 2008, plant procedures will be reviewed and revised as necessary to ensure that all of the new AST operator actions are fully integrated into plant procedures.

2. As described in SNC's RAI response dated February 25, 2008, operators will receive training on the new operator actions associated with AST, including the basis for any new system functions.

2.6.4 <u>Summary</u>

The NRC staff has reviewed the proposed new operator actions to be credited in HNP's safety analysis associated with implementing an AST at HNP, and, as discussed above, finds them to be acceptable.

2.7 <u>Materials and Chemical Engineering</u>

This review section is based on a review of the licensee's submittals dated August 29, 2006, and January 30, 2007, and addresses the portion of the application dealing with the licensee's analysis for maintaining suppression pool pH greater than or equal to 7 for 30 days following a loss of coolant accident (LOCA). According to RG 1.183, maintaining pH basic will prevent re-evolution of iodine from the suppression pool water.

According to NUREG-5950, "Iodine Evolution and pH Control," pH is the major factor in determining the extent of formation of molecular iodine (I^2) in solution during the design basis

LOCA. The value of pH is determined by relative amounts of acidic and basic chemicals introduced to the suppression pool. According to RG 1.183, Appendix A, Section 2, the iodine released from the core is composed of 95% cesium iodide (CsI), 4.85% elemental, and 0.15% organic iodines. CsI is soluble in water and I^2 is scarcely soluble and, when released to the containment atmosphere, may leak to the outside, contributing to the radiation doses. In a radioactive environment existing in the containment, some of the ionic iodine converts into molecular form. The fraction converted depends on the pH of the suppression pool water. At pH values greater than 7, only a negligible amount of ionic iodine is converted into elemental form. Therefore, it is important to maintain pH greater than 7.

The chemicals that principally contribute to acidity in containment are nitric acid (HNO_3), produced by irradiation of water and air, and hydrochloric acid (HCl), produced by irradiation or heating of Hypalon and polyvinyl chloride cable insulation. The radiation field in the suppression pool during a DBA LOCA results from the release of fission products into the pool. The licensee calculated the total amounts of HNO_3 and HCl produced over 30 days to be ~2E-4 mol/L and 5.6E-4 mol/L, respectively.

With the addition of HNO3 and HCl and without buffering action of sodium pentaborate, the pH in the suppression pool water will drop below 7 in about one day. The licensee states that the unbuffered pH of the pool should remain above 7 for at least several hours. The acids added from radiolysis of water and cable is not sufficient to bring the pH below 7 until 1 day after the accident initiation, due to the presence of cesium compounds, which are basic compounds. However, with a sufficient amount of buffer, the pH in the suppression pool could be maintained above 7 for 30 days. Most of the calculations of the suppression pool pH presented in the submittal were performed using a proprietary computer code, STARpH. Although the code itself was not accessible to the NRC staff, it was possible to examine the input to and output from the code. Also, the NRC staff was familiar with the technical basis of the code. This information allowed the NRC staff to evaluate the results generated by the code.

The amount of sodium pentaborate available for injection is 1,975 pounds (lbm) with boron enrichment of 60 percent B10. Although the primary function of the standby liquid control system (SLCS) is to introduce negative reactivity to the core, in the event of a DBA LOCA the SLCS will inject into the pool within several hours of accident initiation for pH control. If the SLCS injects into the suppression pool, substantial mixing will rapidly occur. The licensee states that the significant mixing will occur on the order of 1 hr, based on a residual heat removal flow rate of about 10,000 gpm and pool volume of 7E+05 gallons. Upon SLCS actuation, sodium pentaborate is injected into the pool within approximately 2 hrs of accident initiation. If the reactor vessel does not have an immediate pathway to the suppression pool, this time will be extended by an additional few hours to allow the licensee to assure communication with the pool or inject sodium pentaborate to the pool using another pathway.

The NRC staff concludes that the licensee's proposed actions will maintain the suppression pool pH greater than 7 for 30 days following a LOCA. By maintaining the pH above 7, the fraction of radioactive iodine released into the containment atmosphere is greatly reduced. The methodology described by the licensee to maintain pH above 7 relies on using the buffering action of sodium pentaborate introduced into the suppression pool from the SLCS. The licensee provided supporting documents containing the input and output data from a proprietary computer program called STARpH. The NRC staff has reviewed the analysis and justifications provided by the licensee and concludes that the analysis presented in the licensee's submittal indicates that the suppression pool pH will be maintained at or above 7, over a period of 30 days after a LOCA. Based on the considerations discussed above, the NRC staff finds this to be acceptable.

3.0 Facility Technical Specification (TS) and Operating License Conditions

3.1 Technical Specification (TS) Changes

TS 1.1, Definition of "Dose Equivalent I-131" is discussed in section 2.1.3.

TS 3.4.6, "Reactor Coolant System Specific activity" is discussed in section 2.1.3.

TS 3.6.1.3, SR for secondary containment bypass leakage and SR for MSIV leakage is discussed in section 2.1.3 and with respect to primary containment isolation valves is discussed in section 2.5.1.

TS 3.6.2.5, Residual Heat Removal Drywell Spray, is discussed in section 2.1.3.

TS 3.7.9 for the TB ventilation exhaust system fan is discussed in section 2.5.1.

3.2 License Conditions and Regulatory Commitments

As part of the AST amendment the following items will be included as conditions in the Facility Operating License for HNP Unit 1, as follows:

2.C(9) Alternative Source Term

1) Southern Nuclear shall complete actions by April 30, 2010, as described in Southern Nuclear's letters dated October 18, 2007, and March 13, 2008, to complete the design modifications to the HNP turbine building ventilation exhaust systems. Specifically, the HNP Units 1 and 2 turbine building exhaust fans shall be capable of being manually switched over from normally operating power supplies, to a Class 1E circuit that will be isolated by an appropriately rated safety related, environmentally and seismically qualified circuit breaker. For further protection and isolation, the licensee shall also use fuses that will be located in a seismically qualified manual transfer switch housing. The aforementioned circuit breaker and fuses shall be adequately coordinated with the upstream load center breaker over the entire range. These devices shall be adequately rated to prevent adverse effects of a fault to the rest of the distribution system.

2) Southern Nuclear shall implement modifications by May 31, 2010, as described in Enclosure 1, section 2.7.3.2, of the LAR and section 5.7 of SNC's letter dated February 25, 2008 (NL-08-0175) to modify the design for the air supply to the turbine building exhaust ventilation dampers, such that operating air to the dampers will be supplied from a non-interruptible instrument air source to eliminate single failure point vulnerability to loss of system/instrument air.

3) Southern Nuclear shall complete actions by May 31, 2010, as described in Southern Nuclear's letter dated February 25, 2008 (NL-08-0175) to install and implement the capability for Standby Liquid Control System hand switch jumpers for HNP Units 1 and 2.

4) Southern Nuclear shall complete actions by May 31, 2012, as described in Southern Nuclear's letters dated February 25, 2008 (NL-08-0175) and July 2, 2008 (NL-08-1022), to modify the following Main Steam Isolation Valve ALT boundary valves, such that they can be closed from the main control room even in the event of a loss of offsite power:

   1N38-F101A, 1N38-F101B, 1N33-F012, 1N33-F013

5) Southern Nuclear shall implement actions by May 31, 2010, as described in Southern Nuclear's letter dated February 27, 2008, to assure that temperature switches which monitor charcoal bed temperature meet the environmental qualification requirements of 10 CFR 50.49.

As part of the AST amendment the following items will be included as conditions in the Facility Operating License for HNP Unit 2, as follows:

2.C(3)(g) Alternative Source Term

i) Southern Nuclear shall complete actions by April 30, 2010, as described in Southern Nuclear's letters dated October 18, 2007, and March 13, 2008, to complete the design modifications to the HNP turbine building ventilation exhaust systems. Specifically, the HNP Units 1 and 2 turbine building exhaust fans shall be capable of being manually switched over from normally operating power supplies, to a Class 1E circuit that will be isolated by an appropriately rated safety related, environmentally and seismically qualified circuit breaker. For further protection and isolation, the licensee shall also use fuses that will be located in a seismically qualified manual transfer switch housing. The aforementioned circuit breaker and fuses shall be adequately coordinated with the upstream load center breaker over the entire range. These devices shall be adequately rated to prevent adverse effects of a fault to the rest of the distribution system.

ii) Southern Nuclear shall implement modifications by May 31, 2010, as described in Enclosure 1, section 2.7.3.2, of the LAR and section 5.7 of SNC's letter dated February 25, 2008 (NL-08-0175) to modify the design for the air supply to the turbine building exhaust ventilation dampers, such that operating air to the dampers will be supplied from a non-interruptible instrument air source to eliminate single failure point vulnerability to loss of system/instrument air.

iii) Southern Nuclear shall complete actions by May 31, 2010, as described in Southern Nuclear's letter dated February 25, 2008 (NL-08-0175) to install and implement the capability for Standby Liquid Control System hand switch jumpers for HNP Units 1 and 2.

iv) Southern Nuclear shall complete actions by May 31, 2011, as described in Southern Nuclear's letters dated February 25, 2008 (NL-08-0175) and July 2, 2008 (NL-08-1022), to modify the following Main Steam Isolation Valve ALT boundary valves, such that they can be closed from the main control room even in the event of a loss of offsite power:

2N11-F004A, 2N11-F004B, 2N33-F003, 2N33-F004

v) Southern Nuclear shall implement actions by May 31, 2010, as described in Southern Nuclear's letter dated February 27, 2008, to assure that temperature switches which monitor charcoal bed temperature meet the environmental qualification requirements of 10 CFR 50.49.

License condition 2.C(8) for Unit 1 was revised to extend the date for the crediting of administration of potassium iodide from May 31, 2010, to May 31, 2012, and license condition 2.C(3)(f) for Unit 2 was revised to extend the date from May 31, 2010, to May 31, 2011.

4.0 STATE CONSULTATION

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

5.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The Commission's regulation at 10 CFR 50.92(c) states that the Commission may make a final determination that a license amendment involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) result in a significant reduction in a margin of safety.

The NRC staff reviewed the no significant hazards consideration (NSHC) evaluation provided by the licensee for some of the issues and developed its own evaluation for other issues included in the licensee's application. The NRC staff made a final determination that NSHC is involved for the proposed amendment and that the amendment should be issued as allowed by the criteria contained in 10 CFR 50.91. The NRC staff final determination is presented below:

The proposed amendment includes two actions, as follows. First, the proposed amendment responds to existing license conditions, "Design Bases Accident Radiological Consequences Analyses," by revising the licensing and design basis, including the TS, for four DBAs: the loss-of-coolant, main steamline break, control rod drop and FHAs. Notice of this action was previously published in the *Federal Register* on May 6, 2008 (73 FR 25046). Noticing of this action was again provided on July 23, 2008, (73 FR 42834) to include further supplements to the licensee's August 29, 2006 application, that are dated April 1, May 5, June 25 and July 14, 2008, that were submitted subsequent to the *Federal Register* Notice of May 6, 2008. The July 23, 2008 Notice replaced and superceded the *Federal Register* Notice of May 6, 2008, in its entirety. The second action would be modification of license conditions to extend the implementation date of May 31, 2010 until May 31, 2012 for HNP unit 1 and until May 31, 2011 for HNP unit 2, as discussed in the licensee's letter of July 2, 2008.
The licensee's submittal dated August 14, 2008, was not included within the scope of the NSHC Notice dated July 23, 2008. As also noted in section 1.0 above, the NRC staff finds that the

August 14, 2008, submittal provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission (NRC) staff's proposed no significant hazards consideration determination as published in the *Federal Register* on July 23, 2008 (73 FR 42834)

The Commission has made a final determination that the amendment request involves no significant hazards consideration. Under the Commission's regulations in Title 10 of the CODE OF FEDERAL REGULATIONS (10 CFR), Section 50.92, this means that operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. Based on information as provided in the licensee's submittals for the first action identified above, the NRC staff has determined the following with respect to the three criteria above;

> Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?
>
> Adoption of the AST methodology and allowing credit in the accident analyses for those plant systems affected by implementing AST are not expected to initiate DBAs. The revised accident source term is an input to the radiological consequence analyses. The implementation of the AST and changed TS have been incorporated in the analyses for the limiting DBAs at HNP. The structures, systems, and components affected by the proposed change are mitigative in nature and would be relied upon after an accident has been initiated. Based on the revised analyses, the proposed changes to the TS (including revised leakage limits) impose certain performance criteria on existing systems that do not increase accident initiation probability. The proposed changes do not involve a revision to the parameters or conditions that could contribute to the initiation of a DBA as discussed in Chapter 15 of the Unit 2 FSAR. Therefore, the proposed change does not result in an increase in the probability of an accident previously identified. Plant specific AST radiological analyses have been performed and, based on the results of these analyses, the licensee has demonstrated that the dose consequences of the limiting events considered in the analyses are within the regulatory guidance provided by the NRC for use with the AST as provided in 10 CFR 50.67, Regulatory Guide 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors (ML003716792) and Standard Review Plan, Section 15.0.1. Therefore, the proposed changes do not result in a significant increase in the consequences of an accident previously evaluated.
>
> Does the proposed change create the possibility of a new or different kind of accident from any previously evaluated?
>
> The use of AST methodology and the implementation of limited changes to structures, systems or components (SSC) to support that methodology, does not alter or involve any DBA initiators. No major SSCs are added to or removed from the HNP design. The limited changes in the design of existing SSCs needed to enable crediting their function in currently postulated DBAs and the addition of further TS are intended to enhance the assurance that these SSCs will perform their mitigative function in the event of a DBA.

Since the operation of the SSCs will not be significantly changed after the AST implementation, no new failure modes are created by this proposed change. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

Does the proposed change involve a significant decrease in the margin of safety?

The principal changes in the licensing and design bases for this amendment are associated with demonstrating that the radiological consequences of DBAs meet applicable NRC regulatory criteria, as discussed in criterion 1 above. The licensee states that the analyzed events have been carefully selected, and the analyses supporting these changes have been performed using approved methodologies and conservative inputs to ensure that analyzed events are bounding and safety margin has been retained. The licensee also states that the dose consequences of these limiting events are within the acceptance criteria presented in 10 CFR 50.67, Regulatory Guide 1.183, and Standard Review Plan 15.0.1 and that, because the proposed changes continue to result in dose consequences within the applicable regulatory limits, the changes are considered to not result in a significant reduction in the margin of safety.

For the second issue identified above, the NRC staff has determined the following with respect to the three criteria above:

Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

This proposed change will authorize SNC to credit [potassium iodide] KI for an extended period in the DBA radiological consequences analyses to address the impact of [main control room] MCR unfiltered inleakage. This proposed change does not result in any functional or operational change to any systems, structures, or components and has no impact on any assumed initiator of any analyzed accident. Therefore, the proposed change does not result in an increase in the probability of an accident previously evaluated.

This proposed change does not introduce any additional method of mitigating the thyroid dose to MCR occupants in the event of a loss-of-coolant accident (LOCA) since the existing license condition has already introduced this method as part of the licensing basis for an interim period of time. The updated LOCA MCR radiological dose, considering 110 [cubic feet per minute] cfm unfiltered inleakage and crediting KI, continues to meet GDC 19 acceptance limits. In the context of the current licensing basis with MCR unfiltered inleakage considered, LOCA continues to be the limiting event for radiological exposures to the operators in the MCR. Radiological doses to MCR occupants are within the regulatory limits of GDC 19 with MCR unfiltered inleakage up to 1000 cfm without the crediting of KI for the main steam line break accident (MSLB), CRDA , and FHA. Therefore, the proposed change does not result in a significant increase in the consequences of an accident previously evaluated.

Does the proposed change create the possibility of a new or different kind of accident from any previously evaluated?

This proposed change will authorize SNC to credit KI for an extended period in the DBA radiological consequences analyses to address the impact of MCR unfiltered inleakage. This proposed change does not result in any functional or operational change to any systems, structures, or components. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

Does the proposed change involve a significant decrease in the margin of safety?

This proposed change will authorize SNC to credit KI for an extended period in the DBA radiological consequences analyses to address the impact of MCR unfiltered inleakage. This proposed change does not result in any functional or operational change to any systems, structures, or components. This proposed change extends the use of an additional method of mitigating the thyroid dose to MCR occupants in the event of a LOCA until May 31, 2012, for HNP Unit 1 and until May 31, 2011, for HNP Unit 2. The updated LOCA MCR radiological dose, considering 110 cfm unfiltered inleakage and crediting KI, continues to meet GDC 19 acceptance limits. In the context of the current licensing basis with MCR unfiltered inleakage considered, LOCA continues to be the limiting event for radiological exposures to the operators in the MCR. Radiological doses to MCR occupants are within the regulatory limits of GDC 19 with MCR unfiltered inleakage of up to 1000 cfm without the crediting of KI for the main steam line break accident (MSLB), CRDA , and FHA. Therefore, the proposed change does not involve a significant decrease in the margin of safety.

The NRC staff finds that, on the basis discussed above, that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff determines that the amendment request involves no significant hazards consideration.

6.0 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, and changes surveillance requirements. 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 NRC staff has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (73 FR 42834, July 23, 2008). 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.

7.0 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 amendments will not be inimical to the common defense and security or to the health and safety of the public.

References:

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2. SNC letter NL-06-2496, dated November 6, 2006, Ml063170194, on atmospheric dispersion factors.

3. SNC letter NL-06-2550, dated November 27, 2006, ML063380180, on atmospheric dispersion factors.

4. SNC letter NL-06-2750, dated January 30, 2007, ML070330404, on suppression pool pH control analysis

5. SNC letter dated June 22, 2007, ML071770585, Response to Request for Additional Information Regarding the Unit 1 Main Steam Isolation Valve Alternate Leakage Treatment Path Evaluation.

6 SNC letter NL-07-1308, dated July 16, 2007, ML071970469, response to RAI regarding the unit 1 main steam isolation valve alternate leakage treatment path seismic evaluation

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18. SNC submittal, dated June 25, 2008, responding to RAI on technical specifications, SNC letter NL-08-0903, ML081780683.

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20 Letter with enclosures, J.T. Beckham, Jr., V.P. - Nuclear, Hatch Project, Georgia Power to USNRC, Subject: Edwin I. Hatch Nuclear Plant – Unit 2 Request to Revise Technical Specifications: Increase in Allowable MSIV Leakage Rate and Deletion of the MSIV Leakage Control System, Docket No. 50-366, October 1, 1993.

21 Letter with enclosures, J.T. Beckham, Jr., V.P. – Nuclear, Hatch Project, Georgia Power to USNRC, Subject: Response to Request for Additional Information: Edwin I. Hatch Nuclear Plant Request to Revise Technical Specifications - Increase in Allowable MSIV Leakage Rate and Deletion of the MSIV Leakage Control System, Docket No. 50-366, February 3, 1994.

22 Letter with enclosures, Kahtan N. Jabbour, Project Manager, USNRC Office of Nuclear Reactor Regulation to J.T. Beckham, Jr., Vice President – Plant Hatch, Georgia Power Company, Subject: Issuance of Amendment – Edwin I. Hatch Nuclear Plant, Unit 2 (TAC No. M87850), March 17, 1994.

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ACRONYMS

ADAMS	Agencywide Documents Access and Management System
AST	Alternate Source Term
CRDA	Control Rod Drop Accident
DBA	Design Basis Accident
DF	Decontamination Factor
EAB	Exclusion Area Boundary
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
ESF	Engineered Safety Feature
FHA	Fuel Handling Accident
FSAR	Final Safety Analysis Report
GDC	General Design Criteria
HNP	Edwin I. Hatch Nuclear Plant, Units 1 and 2
ISI	Inservice Inspection
IST	Inservice Testing
KI	Potassium Iodide
LAR	License Amendment Request
LCO	Limiting Condition of Operation
LOCA	Loss of Coolant Accident
LOOP	Loss of Offsite Power
LPZ	Low Population Zone
MCR	Main Control Room
MSIV	Main Steam Isolation Valve
MSL	Main Steam Line
MSLB	Main Steam Line Break
MWd/MTU	megawatt days per metric ton of uranium
NRC	U. S. Nuclear Regulatory Commission
PBWS	Banked Position Withdrawal Sequence
PC	Primary Containment
RG	Regulatory Guide
SE	Safety Evaluation
SGTS	Standby Gas Treatment System
SLCS	Standby Liquid Control System
SNC	Southern Nuclear Operating Company
SPB	Sodium Pentaborate
SRP	Standard Review Plan
SSCs	Structures, Systems and Components
SSE	Safe Shutdown Earthquake
TB	Turbine Building
TEDE	Total Effective Dose Equivalent
TID	Technical Information Document
TS	Technical Specifications
TSC	Technical Support Center