June 15, 2007

Mr. Thomas D. Walt, Vice President
Carolina Power & Light Company
H. B. Robinson Steam Electric Plant
Unit No. 2
3581 West Entrance Road
Hartsville, South Carolina 29550

SUBJECT: H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2 — ISSUANCE OF AN AMENDMENT FOR TECHNICAL SPECIFICATIONS CHANGES RELATED TO CONTAINMENT PEAK PRESSURE (TAC NO. MD2682)

Dear Mr. Walt:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 215 to Renewed Facility Operating License No. DPR-23 for the H. B. Robinson Steam Electric Plant, Unit No. 2. This amendment changes the Technical Specifications (TS) in response to your application dated July 17, 2006. Specifically, the amendment revises the containment design pressure in Surveillance Requirement 3.6.8.1 and 3.6.8.5 concerning the "Isolation Valve Seal Water System," and TS Section 5.5.16 "Containment Leakage Rate Testing Program."

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

Sincerely,

/RA/

Chandu P. Patel, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-261

Enclosures:
1. Amendment No. 215 to DPR-23
2. Safety Evaluation

cc w/encls: See next page

June 15, 2007

Mr. Thomas D. Walt, Vice President
Carolina Power & Light Company
H. B. Robinson Steam Electric Plant
Unit No. 2
3581 West Entrance Road
Hartsville, South Carolina 29550

SUBJECT: H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2 — ISSUANCE OF AN AMENDMENT FOR TECHNICAL SPECIFICATIONS CHANGES RELATED TO CONTAINMENT PEAK PRESSURE (TAC NO. MD2682)

Dear Mr. Walt:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 215 to Renewed Facility Operating License No. DPR-23 for the H. B. Robinson Steam Electric Plant, Unit No. 2. This amendment changes the Technical Specifications (TS) in response to your application dated July 17, 2006. Specifically, the amendment revises the containment design pressure in Surveillance Requirement 3.6.8.1 and 3.6.8.5 concerning the "Isolation Valve Seal Water System," and TS Section 5.5.16 "Containment Leakage Rate Testing Program."

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

Sincerely,

/RA/

Chandu P. Patel, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-261

Enclosures:
1. Amendment No. 215 to DPR-23
2. Safety Evaluation

cc w/encls: See next page

DISTRIBUTION:

PUBLIC	RidsNrrLACSola	RidsNrrDssScvb	R. Lobel, NRR
Lpl2-2 Rdg.	RidsOgcRp	RidsRgn2MailCenter	R. Goel, NRR
RidsNrrDorlLpl2-2	G. Hill, OIS (2)	RidsAcrsAcnwMailCenter	
RidsNrrPMCPatel	RidsNrrDorlDpr	L. Regner, NRR	

ADAMS Accession No. ML071070170 Package: ML071720033 TS: ML071720254

OFFICE	LPL2-2/PM	LPL2-2/PM	LPL2-2/LA	SCVB/BC	OGC	LPL2-2/BC
NAME	LRegner	CPatel	CSola	RDennig by memo dated		TBoyce
DATE	4 /17 /07	4/17 /07	4/17 /07	3/26/07	06/08/07	06/15/07

OFFICIAL RECORD COPY

cc:

Mr. David T. Conley
Associate General Counsel II - Legal
Department
Progress Energy Service Company, LLC
Post Office Box 1551
Raleigh, North Carolina 27602-1551

Ms. Margaret A. Force
Assistant Attorney General
State of North Carolina
Post Office Box 629
Raleigh, North Carolina 27602

U. S. Nuclear Regulatory Commission
Resident Inspector's Office
H. B. Robinson Steam Electric Plant
2112 Old Camden Road
Hartsville, South Carolina 29550

Mr. Ernest J. Kapopoulos, Jr.
Plant General Manager
H. B. Robinson Steam Electric Plant,
Unit No. 2
Carolina Power & Light Company
3581 West Entrance Road
Hartsville, South Carolina 29550

Public Service Commission
State of South Carolina
Post Office Drawer 11649
Columbia, South Carolina 29211

Mrs. J. F. Lucas
Manager - Support Services - Nuclear
H. B. Robinson Steam Electric Plant,
Unit No. 2
Carolina Power & Light Company
3581 West Entrance Road
Hartsville, South Carolina 29550

Mr. C. T. Baucom
Supervisor, Licensing/Regulatory Programs
H. B. Robinson Steam Electric Plant,
Unit No. 2
Carolina Power & Light Company
3581 West Entrance Road
Hartsville, South Carolina 29550

Ms. Beverly Hall, Section Chief
N.C. Department of Environment
and Natural Resources
Division of Radiation Protection
3825 Barrett Dr.
Raleigh, North Carolina 27609-7721

Mr. Robert P. Gruber
Executive Director
Public Staff - NCUC
4326 Mail Service Center
Raleigh, North Carolina 27699-4326

Mr. Henry H. Porter, Assistant Director
South Carolina Department of Health
Bureau of Land & Waste Management
2600 Bull Street
Columbia, South Carolina 29201

Mr. J. Paul Fulford
Manager, Performance Evaluation and
Regulatory Affairs PEB 5
Carolina Power & Light Company
Post Office Box 1551
Raleigh, North Carolina 27602-1551

Mr. John H. O'Neill, Jr.
Shaw, Pittman, Potts, & Trowbridge
2300 N Street NW.
Washington, DC 20037-1128

CAROLINA POWER & LIGHT COMPANY

DOCKET NO. 50-261

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

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 215
Renewed License No. DPR-23

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

   A. The application for amendment by Carolina Power & Light Company (the licensee), dated July 17, 2006, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications, as indicated in the attachment to this license amendment; and paragraph 3.B. of Renewed Facility Operating License No. DPR-23 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 215, are hereby incorporated in the license.

   The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Thomas H. Boyce, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Operating License No. DPR-23
and the Technical Specifications

Date of Issuance: June 15, 2007

ATTACHMENT TO LICENSE AMENDMENT NO. 215

RENEWED FACILITY OPERATING LICENSE NO. DPR-23

DOCKET NO. 50-261

Replace page 3 of Operating License No. DPR-23 with the attached page 3.

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

Remove Page	Insert Pages
3.6-20	3.6-20
3.6-21	3.6-21
5.0-24	5.0-24

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 215 TO

RENEWED FACILITY OPERATING LICENSE NO. DPR-23

CAROLINA POWER & LIGHT COMPANY

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

DOCKET NO. 50-261

1.0 INTRODUCTION

By letter dated July 17, 2006, the Carolina Power & Light Company (licensee) submitted a request for changes to the H. B. Robinson Steam Electric Plant, Unit No. 2 (HBR), Technical Specifications (TS). The requested changes involve revisions to containment peak pressure as a result of a revised loss-of-coolant accident (LOCA) containment pressure analysis. The revised analysis calculated a peak containment pressure (designated as P_a in the TS) following an LOCA of 41.49 pounds per square in gauge (psig), which is greater than the current TS value for P_a of 40.5 psig. The change defines P_a as the containment design pressure of 42 psig, which is conservative compared to the revised post-LOCA peak pressure of 41.49 psig.

The TS sections affected are:

1. TS Section 3.6.8, “Isolation Valve Seal Water System”

   Surveillance Requirements (SR) 3.6.8.1 and 3.6.8.5 specify a pressure of 44.6 psig. This value is based on 1.1 times the existing P_a of 40.5 psig. Since P_a is being revised to 42 psig, the value in the two SR are being increased to 46.2 psig.

2. TS Section 5.5.16, “Containment Leakage Rate Testing Program”

   This section defines P_a as the peak calculated containment internal pressure for the design basis LOCA and specifies a value of 40.5 psig. It is being revised to specify P_a as the containment design pressure of 42 psig.

The licensee indicated that the change was necessary due to a revision of the post-LOCA containment analysis. The licensee requested Westinghouse to recalculate the containment analysis due to non-conservatisms discovered in the current analysis concerning the mass and energy released to the containment following a postulated LOCA.

The licensee indicated that the revised analysis does not require changes to the existing surveillance procedures. Current surveillances of containment leakage (both integrated leakage rate testing and local leakage rate testing) are performed at pressures in excess of 42 psig. Current surveillances of the Isolation Valve Seal Water System have been performed at pressures in excess of 46.2 psig. Therefore, the current plant procedures and current plant conditions are consistent with the proposed change. The proposed change will ensure the TS are consistent with the corrected analysis.

2.0 REGULATORY EVALUATION

Containment integrity analysis is performed to ensure that the pressure and temperature inside containment will remain below the containment building design conditions if a high-energy line break inside containment should occur during plant operation. The analysis ensures that the containment heat removal capability is sufficient to remove the maximum possible discharge of mass and energy to containment from the Nuclear Steam Supply System without exceeding the acceptable criteria (design pressure and temperature).

The relevant regulatory requirements for acceptance of the proposed change are contained in the HBR Updated Final Safety Analysis Report (UFSAR) as it relates to containment design integrity and containment heat removal and in Title 10 to the *Code of Federal Regulations* (10 CFR) Part 50, Appendix J, *Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors*.

General Design Criteria (GDC) 10, 49, and 52 as discussed in Chapter 3.1, *Conformance with General Design Criteria*, of HBR's UFSAR relate to containment design integrity and containment heat removal. However, the number designation for HBR GDC differ from the current 10 CFR Part 50 Appendix A GDC. This is due to the fact that the GDC in existence at the time HBR was licensed (July 1970) for operation were contained in Proposed Appendix A to 10 CFR Part 50, *General Design Criteria for Nuclear Power Plants* published in the *Federal Register* on July 11, 1967. Appendix A to 10 CFR Part 50, effective in 1971 and subsequently amended, is different from the proposed 1967 criteria. Future discussions in this Safety Evaluation (SE) refer to the proposed GDC contained in HBR's UFSAR.

Proposed GDC 10, in part, refers to design of the containment to withstand a large reactor coolant system pipe break without loss of integrity, and, together with other engineered safety features, retain its functional capability.

Proposed GDC 49, in part, refers to design of the containment structure to limit leakage of radioactive materials from openings and penetrations and any necessary heat removal systems, so as not to result in undue risk to the health and safety of the public.

Proposed GDC 52, in part, refers to the design of active heat removal systems such that they prevent exceeding containment design pressure under accident conditions and shall perform their required function assuming a single failure of an active component.

The licensee has evaluated the proposed change against the applicable Nuclear Regulatory Commission (NRC) regulations and criteria.

3.0 TECHNICAL EVALUATION

The design and licensing of nuclear power plants require that the containment be analyzed for pressure and temperature effects of postulated accidents to ensure that design limits are not exceeded. The analyses include pressure and temperature transients to which the containment might be exposed as a result of postulated reactor coolant system pipe breaks. Containment integrity analyses are performed to quantify the margin in the containment design pressure and in the peak temperature and pressure for equipment environmental qualification (EQ), and to demonstrate the acceptability of the containment safeguards equipment to mitigate the postulated LOCA.

The licensee requested that Westinghouse reanalyze the containment analysis due to non-conservatisms discovered in the current analysis. The non-conservatisms only impacted the LOCA analysis and not the main steam line break analysis. The Westinghouse report presented revised mass and energy releases for the postulated LOCA accident due to Westinghouse identified issues with respect to the current analysis. Eight issues were identified which may impact the current LOCA mass and energy release analysis and, therefore, calculated peak containment pressure. These issues are discussed below:

3.1 Issues Affecting Mass and Energy Release Calculations

3.1.1 Area of the downcomer in the REFLOOD computer code

Westinghouse-designed reactors can be divided into downflow and upflow barrel baffle designs; HBR is a downflow plant. During a Westinghouse review of the values used for the downcomer area for downflow plants, the values were determined to be incorrect. A larger downcomer area and thus a larger volume for downflow plants was determined to be correct. Consequently, a longer time was required for the emergency core cooling system (ECCS) to completely fill the downcomer. The calculation was performed with the Westinghouse REFLOOD computer code presented in WCAP-10325P-A *Westinghouse LOCA Mass and Energy Release Model for Containment Design* (the WCAP Topical) which was found acceptable by the NRC in February 1987. The licensee stated that sensitivity studies showed that the effect on LOCA mass and energy release, as determined by the resulting effect on the containment pressure, was small and can be neglected. Based on the results of the Westinghouse calculations, the revised analysis of the area of the downcomer assumed in the REFLOOD calculations is acceptable.

3.1.2 Area of the upper plenum in the FROTH code

The FROTH computer program is run in conjunction with the REFLOOD computer program and calculates the LOCA mass and energy releases for the post-reflood period until the steam generator (SG) secondary side pressure is calculated to equilibrate at the containment design pressure. During this time period, the two-phase mixture levels in the core, upper plenum, hot leg and SG inlet plenum are the principal parameters of interest. Westinghouse determined that in certain instances the cross-sectional area of the upper plenum was being over-predicted, which resulted in a reduction in entrainment to the SGs and thus less steam production. Correction of the upper plenum area results in increased mass and energy releases and a penalty for the calculated containment pressure. The revised analyses include this correction and are acceptable to the NRC staff. The FROTH code has previously been approved by the NRC staff in the WCAP Topical.

3.1.3 Definition for other FROTH inputs

A Westinghouse review of the FROTH code input variables showed that the SG inlet plenum flow area, which is used to calculate the void fraction in the SG inlet plenum, was based on a value that was too small. This review of the SG geometry for the SG inlet plena determined a more appropriate method for calculating SG inlet plenum flow area. The result was a larger flow area and a reduction in entrained liquid. The reduction in entrained liquid reduces the mass and energy released post-reflood, and is a benefit for the calculated containment pressure. Since the new calculation of the SG inlet plenum flow area is more representative of the actual design, the NRC staff finds this correction to be acceptable.

3.1.4 Commitments made within WCAP-10325 and the WCAP 10325-P Safety Evaluations

The NRC staff approved the WCAP Topical in February 1987. This model has been used to calculate the LOCA mass and energy releases for almost all Westinghouse-designed pressurized water reactors (PWRs) and some non-Westinghouse designed PWRs. As a result of a Westinghouse review of these models, the need to clarify two model features was identified. These features are: (1) the assumption of the SG exit steam enthalpy during the post-reflood period, and (2) the assumed power level used in the LOCA mass and energy release analysis. The differences in the models have been determined to be very small relative to the overall conservatism of the analysis. However, since these two model features are applied differently than approved, Westinghouse deemed it necessary to report these discrepancies to the NRC staff. The NRC's SE addressing these issues is contained in a letter dated October 18, 2005. The SE determined that no changes to the current model described in WCAP Topical are required.

3.1.5 Main feedwater addition following a reactor trip

The WCAP Topical indicated that Westinghouse needed to account for the addition of main feedwater (MFW) to the secondary side of the SGs following an LOCA in the time from reactor trip until MFW isolation is calculated to occur. The recent methodology review called into question the current modeling of the isolation of the SG secondary side on a reactor trip signal. The continued addition of MFW after a reactor trip would add energy to the secondary side above 212 degrees Fahrenheit (°F). This additional energy would be released to the containment. Depending upon the time at which peak containment pressure is calculated, Westinghouse sensitivity studies have shown that the peak pressure may be underestimated. Since the change more accurately accounts for a source of energy and more accurately reflects the system response to the LOCA, the NRC staff finds this correction to be acceptable.

3.1.6 Considerations for auxiliary feedwater (AFW) system purge and unisolable volumes

After MFW isolation, a volume of hot MFW resides in the main feed lines between the AFW injection point and the SG secondary side. Once AFW flow is initiated, the hot MFW water would be pushed into the SG secondary side. Since the SGs are calculated to depressurize, there may be additional volume trapped between the AFW injection point and the MFW isolation valve that would flash and be pushed into the SG secondary. These two concerns were not considered in Robinson's FSAR and the WCAP Topical LOCA mass and energy release models. Addition of these effects to the LOCA mass and energy release calculation has been shown by Westinghouse calculations to increase the total energy released to containment and results in an increase in the calculated containment pressure. The licensee did not model the AFW purge and unisolable volumes in the containment LOCA analysis. Instead, the licensee assumed no credit for cold AFW. This is more conservative and is therefore acceptable to the NRC staff.

3.1.7 Inadequate definition of required AFW flow for the FROTH code

In recent Westinghouse data requests to licensees to support new LOCA mass and energy release analyses, post-LOCA minimum AFW flow is one of the items requested. In some cases, the actual flow used in the analysis assumed that the flow provided was per SG, instead of the total flow to all SGs. This resulted in flow that was high, and therefore a non-conservatively low energy release to containment. For plants that currently do not credit any AFW flow during the post-reflood period, there is no impact on the analysis results. For plants that credit too large an AFW flow, this results in a decrease in the calculated containment pressure. The licensee used the more conservative approach by not crediting the cold AFW flow. This is acceptable to the NRC staff.

3.1.8 Possibility of asymmetric AFW flow

LOCA analyses are performed assuming a loss of off-site power and the failure of one emergency diesel generator. If the plant is not designed to start the turbine-driven AFW pump on the loss of offsite power or an ECCS signal, then this results in one motor-driven AFW pump in operation under the above conditions. One AFW pump will not feed all SGs; thus, one or more SGs may receive reduced or no AFW flow. In some cases, the flow used for the analysis assumed full flow to each SG which would not be true if only one AFW pump is in operation. The assumption of full AFW flow results in a non-conservative lower energy release to containment. The current LOCA mass and energy release models do not contain a provision to model asymmetric AFW flow (unequally distributed flow to the SGs). Instead, this effect is bounded by the assumption of no AFW flow to the SGs. The licensee stated that it is possible for HBR to have asymmetric flow; therefore, the licensee did not credit the AFW flow in the new analysis. This is conservative and therefore acceptable to the NRC staff.

3.1.9 Additional Changes not reported in the Analysis Report

The licensee indicated that other changes to the HBR LOCA mass and energy input model were made during the new analysis but were not reported in the Analysis Report. The licensee stated that these changes have little effect on the results and in some instances result in lower calculated peak pressure. These changes are the SG outlet nozzle hydraulic diameter and flow area values and reactor coolant pump related data used in the Input Modification Program. The NRC staff concurs with the licensee's findings.

The licensee's submittal describes the Input Modification Program as a data base of generically applicable data used to develop a plant specific input model. The licensee identified some errors in the data and made corrections for the new analysis. Since these changes make the model more accurate, the NRC staff finds these changes to be acceptable.

3.1.10 Break Location

The licensee stated that only the Double Ended Pump Suction (DEPS) break with minimum ECCS flow needs to be reanalyzed. The Double Ended Hot Leg (DEHL) break which calculates a peak pressure to occur during blowdown could only be affected by the post-LOCA addition of MFW. Since MFW would cool the SG side during blowdown, the transfer of heat from secondary to primary would be reduced resulting in less energy released. Thus, the DEHL break is considered to be unaffected by the reported issues. The NRC staff concurs.

The DEPS break with maximum ECCS flow, while affected, has not been reanalyzed since the results for this break are well below the DEPS case with minimum ECCS flows. In a May 27, 1999, letter, the licensee states that the DEPS maximum ECCS case was 2.33 psi below the DEPS minimum ECCS case. This delta is greater than the sum of the effects seen for the DEPS minimum ECCS case. Thus, a reanalysis of the DEPS maximum ECCS case is not expected to exceed the results calculated for the DEPS minimum ECCS case.

3.2 Containment Analysis

3.2.1 LOCA Mass and Energy Release Calculations

Attachment 2 to the licensee's July 17, 2006, letter describes the methods used to determine the mass and energy release rates to the containment following an LOCA and the containment pressure and temperature calculations. The evaluation model used for the long-term LOCA mass and energy release calculation is the March 1979 model described in the WCAP Topical and approved by NRC letter dated October 18, 2005. This evaluation model has been reviewed and approved generically by the NRC staff and has been utilized for other Westinghouse PWRs.

A total of three LOCA mass and energy release cases are presented in the Westinghouse analysis report. The cases addressed two different break locations, the DEHL break and the DEPS break, with each location analyzed for both minimum and maximum pumped ECCS flows. The minimum ECCS cases were performed to address maximum available steam release (minimizing steam condensation). The maximum ECCS cases were performed to address the effects of maximum mass flow and the subsequent effect on containment response. The WCAP Topical provided justification that these analyses encompass the most limiting assumptions for break location and safeguards operation. Only the DEPS break with the assumption of minimum ECCS flow has been reanalyzed as discussed in Section 3.1.10 of this evaluation.

The mass and energy release analysis is sensitive to the assumed characteristics of various plant systems; some of the most critical items are the RCS initial conditions, core decay heat, accumulators, ECCS flow, primary and secondary metal mass heat release modeling, and SG heat release modeling. All input parameters are determined based on NRC-accepted methodology. The American Nuclear Society (ANS) Standard 5.1 decay heat model has been applied to the fission product decay heat as specified in ANS Institute/ANS-5.1 1979 *American National Standard for Decay Heat Power in Light Water Reactors* dated August 29, 1979 and NRC Information Notice 96-39: *Estimates of Decay Heat Using ANS 5.1 Decay Heat Standard May Vary Significantly* dated July 5, 1996. Input parameters and mass and energy releases are presented in Tables 1 through 19 of the licensee's submitted Westinghouse report.

The consideration of the various energy sources in the long-term mass and energy release analysis provides assurance that all available sources of energy have been included in this analysis. Thus, the review guidelines presented in Standard Review Plan Section 6.2.1.3 have been satisfied.

There are no acceptance criteria for the generation of mass and energy releases. The acceptance criteria are associated with the containment integrity analyses.

3.2.2 LOCA Containment Response (COCO) Analysis

The containment response analysis used the results of the long term mass and energy release calculations discussed in previous sections. Calculation of containment pressure and temperature is accomplished by the COCO computer code discussed in WCAP-8327 *Containment Pressure Analysis Code*. The NRC staff has found the COCO code to be acceptable to calculate the containment response to LOCAs for many nuclear power plants including HBR.

The containment response analysis demonstrates the acceptability of the containment safeguards systems to mitigate the consequences of an LOCA inside containment given the mass and energy releases calculated as discussed previously. In support of equipment design and licensing criteria (e.g., qualified operating life), with respect to post-accident environmental conditions, long term containment pressure and temperature transients are generated to conservatively bound the potential post-LOCA containment conditions.

An analysis of containment response to the rupture of the RCS starts with knowledge of the initial conditions in the containment. All input parameters and assumptions of the initial conditions in the containment (e.g., pressure, temperature, relative humidity, and cooling systems performance) are chosen conservatively, based on accepted methodology, as shown in Table 20 of Attachment 2 to the licensee's July 17, 2006, letter. The NRC staff finds the licensee's analysis acceptable.

3.2.3 Containment Analysis Results

The containment pressure, steam temperature and water (sump) temperature profiles from each of the LOCA cases are shown in the licensee's July 17, 2006, letter for the DEPS and the DEHL break cases. The DEPS case with maximum ECCS flows and the DEHL case were not reanalyzed since they are not limiting, and therefore, the results are identical to the results in the May 27, 1999, letter.

The DEPS LOCA with minimum ECCS flows has been reanalyzed to address the issues reported by Westinghouse. The peak pressure for the new DEPS break with minimum ECCS flow was calculated to be 41.49 psig, which is less than 42 psig design pressure that the licensee has chosen as the new value of P_a. The containment pressure is below 50 percent of the peak value within 24 hours, which satisfies the guidelines of Standard Review Plan Section 6.2.1.1.A. The EQ case for the DEPS break with minimum ECCS flow result of 263.73°F exceeds the previous result of 261.76°F. The licensee determined that the increase in calculated peak temperature of 2°F has no impact on structures or equipment qualification. The NRC staff concurs with the licensee's determination.

Since all containment acceptance criteria continue to be satisfied and conservative and acceptable methods were used in the mass and energy and containment analyses, the staff concludes that HBR complies with HBR GDC 10, 49, and 52.

4.0 CONCLUSION

The staff has reviewed the licensee's LOCA containment analyses containing corrections to input identified by Westinghouse and finds the licensee's analysis methods and results acceptable. Since the analytic methods are acceptable and the corrections to certain input parameters are acceptable, the staff concludes that the licensee is in compliance with HBR GDC 10, 49, and 52. In addition, based on the above evaluation, the staff concludes that the proposed technical specifications changes are acceptable.

5.0 STATE CONSULTATION

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

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 Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (71 FR 51225). 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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: Raj Goel
Richard Lobel

Date: June 15, 2007