Progress Energy

10 CFR 50.90

Serial: RNP-RA/03-0121

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United States Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261/LICENSE NO. DPR-23

SUPPLEMENT TO AMENDMENT REQUEST REGARDING
ONE-TIME EXTENSION OF CONTAINMENT TYPE A TEST INTERVAL

Ladies and Gentlemen:

By letters dated June 11, 2003 and August 20, 2003, Progress Energy Carolinas, Inc., submitted a request for Technical Specifications change regarding a one-time extension to the containment Type A leak rate test interval for the H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2.

Requests for Additional Information (RAIs) related to this change were received from the NRC in a faxed correspondence dated September 17, 2003. Attachment II provides the responses to the RAIs. The responses do not impact the proposed Technical Specifications, No Significant Hazards Consideration Determination, or Environmental Impact Consideration provided in the previous submittals.

Attachment I provides an Affirmation pursuant to 10 CFR 50.30(b).

In accordance with 10 CFR 50.91(b), the State of South Carolina is being provided a copy of this letter.

If you have any questions concerning this matter, please contact Mr. C. T. Baucom.

Sincerely,

Jan F. Lucas
Manager - Support Services - Nuclear

Progress Energy Carolinas, Inc.
Robinson Nuclear Plant
3581 West Entrance Road
Hartsville, SC 29550

A017I

Attachments:

I. Affirmation

II. Responses to NRC Requests for Additional Information

RAC/rac

c: Mr. T. P. O'Kelley, Director, Bureau of Radiological Health (SC)
Mr. H. J. Porter, Director, Division of Radioactive Waste Management (SC)
Mr. L. A. Reyes, NRC, Region II
Mr. C. P. Patel, NRC, NRR
NRC Resident Inspectors, HBRSEP
Attorney General (SC)

AFFIRMATION

The information contained in letter RNP-RA/03-0121 is true and correct to the best of my information, knowledge, and belief; and the sources of my information are officers, employees, contractors, and agents of Progress Energy Carolinas, Inc. I declare under penalty of perjury that the foregoing is true and correct.

Executed On: 13 October 2003

J. W. Moyer
Vice President, HBRSEP, Unit No. 2

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

RESPONSES TO NRC REQUESTS FOR ADDITIONAL INFORMATION

Question 1

Risk information submitted as a part of the severe accident mitigation alternative analysis for Robinson license renewal indicated a population dose of 10.68 person-rem per year for Robinson. In contrast, information submitted as a part of the Integrated Leak Rate Test (ILRT) extension request (e.g., Table 5 of Calculation RNP-F/PSA-0020) shows a population dose of 90.6 person-rem per year based on the same PRA version (MOR99). Please explain the reasons for this disparity and reconcile the differences. Provide an updated analysis as appropriate.

Response 1

The H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2, model of record, referred to as MOR99, does not include "level 3" population dose information. An estimate of population dose-risk was prepared for HBRSEP, Unit No. 2, by a consultant in support of the HBRSEP, Unit No. 2, License Renewal Severe Accident Mitigation Alternatives (SAMA) evaluation. For the ILRT extension risk evaluation, a different method, developed by another consultant, was used to estimate population dose-risk.

The method used in the SAMA dose evaluation is described in Appendix F to the HBRSEP, Unit No. 2, License Renewal Application, dated June 14, 2002. This approach used the MACCS2 code and was based on detailed modeling of plant specific population information, meteorology, evacuation details, and other plant specific information.

As presented in report RSC 01-44, previously supplied by HBRSEP, Unit No. 2, letter dated March 26, 2002, the approach used to estimate doses in support of the ILRT extension request was based on a regression of dose on release fractions, using a set of data from published PRAs and other documents. This was true for all release categories except intact containment, for which a separate detailed plant specific evaluation was performed. The RSC 01-44 approach determined an equation that may be used to calculate dose for a plant, given information about release fractions. This approach implicitly considers variables such as population distribution and evacuation, but does not explicitly evaluate them. This may result in a conservative estimation of doses for HBRSEP, Unit No. 2, which has a relatively low population density. Also, as discussed in report RSC 01-44, doses from noble gas releases may be overpredicted for lower leakage rates using the RSC 01-44 method.

Guidance for approaches to assessing the dose-risk of an ILRT extension was provided in a letter from the Nuclear Energy Institute (A. Pietrangelo, November 13, 2001). This guidance indicates that detailed plant specific level 3 results are not required for evaluation of risks due to an ILRT extension, and further indicates that either information from NUREG-1150 or plant specific information may be used. The results obtained using the RSC 01-44 method appear to be

generally consistent with results obtained for plants in the NUREG-1150 analysis series. The results are plant specific to a degree, since specific release fraction information from HBRSEP, Unit No. 2, was utilized. However, the results are shown to be conservative as compared to a method that uses additional plant specific information, such as population density. Therefore, the results obtained are appropriate for the current ILRT extension request.

Question 2

Inspections from some reinforced and steel containments (e.g., North Anna, Brunswick, D. C. Cook, and Oyster Creek) have indicated degradation from the uninspectable (embedded) side of the steel shell and liner of primary containments. Please describe the uninspectable areas of the Robinson containment, and the programs used to monitor their condition. Provide a quantitative assessment of the impact on LERF due to age-related degradation in these areas, in support of the requested ILRT interval extension to 15 years.

Response 2

Containment Inspection Description

The HBRSEP, Unit No. 2, reactor containment structure is a steel lined concrete shell in the form of a vertical right cylinder with a hemispherical dome and a flat base supported by means of piles. The structure consists of sidewalls measuring 126 feet from the liner on the base to the spring line of the dome, and an inside diameter of 130 feet. The containment liner is designed as a leak proof membrane and is not relied upon for the structural integrity of the containment, except for resisting tangential shears in the dome. It is anchored to the concrete by means of "KSM" shaped steel studs. The liner is not anchored to the concrete base slab and hence does not act compositely with it. The cylindrical portion of the liner and a limited section of the dome liner are insulated. The face of the liner plate in contact with the concrete has no primer or paint applied; the intimate contact with the concrete provides corrosion protection. Beneath the bottom panels of insulation is a moisture barrier, designed to prevent water from reaching the area where the containment liner contacts the concrete base slab. The design of the containment structure is discussed in HBRSEP, Unit No. 2, Updated Final Safety Analysis Report, Section 3.8.1.

The HBRSEP, Unit No. 2, letter dated March 26, 2002, described nine requests for relief from certain requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, involving containment inspections. Of these nine relief requests, two involve inspections of the containment liner and moisture barrier. Relief Request IWE/IWL-01, approved by the NRC in a Safety Evaluation dated July 26, 1999, provided relief from performing a VT-3 visual examination of 100% of the accessible surface areas of containment. The removal and reinstallation of the insulation sheathing panels was determined to be time consuming and result in hardship and unusual difficulty. The alternative authorized by the NRC is to perform a VT-3 visual examination on those portions of the insulated containment liner that are exposed when a maintenance activity requires removal of the liner insulation. Relief Request IWE/IWL-02 authorized a similar alternative for visual inspection of the containment moisture barrier in the Safety Evaluation dated July 26, 1999.

Three areas of the containment vessel liner are typically not accessible and therefore not routinely examined:

- The area below the concrete at elevation 228 inside containment. This is primarily a horizontal surface that is embedded in the concrete. Vertically, this portion of the containment vessel liner extends to elevation 228 and has not been examined.
- The area behind the insulation on the vertical portion inside containment. This portion is behind insulation and sheathing panels and extends from elevation 228 to 367 ft 10 inches. Approximately 100 panels (3'-8" x 7'-8") have been removed for maintenance purposes and the containment vessel liner has been examined and determined to meet design thickness.
- The area on the exterior side of the containment vessel liner. This area includes 100% of the entire portion of the containment vessel liner. Although this embedded side has not been examined, ultrasonic measurements have been taken for the approximately 100 insulation and sheathing panels that have been removed. These ultrasonic measurements have not indicated any degradation from the embedded side of the containment vessel liner.

The following table provides a history of IWE and IWL examinations since the last Type A containment leak rate test:

Outage	Examinations Performed
RO-18 (1998)	Portions of the containment vessel liner behind the insulation.
RO-19 (1999)	Portions of the containment vessel liner behind the insulation, electrical penetrations, airlock, and portions of the reinforced concrete exterior.
RO-20 (2001)	Portions of the containment vessel liner behind the insulation, the dome interior, mechanical penetrations, equipment hatch, and the remaining portions of the reinforced concrete exterior, including the dome exterior.

The conclusions of these IWE and IWL examinations, which were completed in May 2001 (RO-20), provide assurance that the structural integrity and leak-tightness of the HBRSEP, Unit No. 2, containment have not been compromised. Continued performance of these examinations provides a high degree of assurance that any degradation of the containment structure will be detected and corrected before it can produce a containment leakage path or impact structural integrity. For example, during RO-22, an additional five and one-half insulation panels are planned to be removed from the bottom row to allow examination of the containment vessel liner behind the insulation. This will complete examination of the containment vessel liner for the first row of insulation panels around the entire perimeter of containment interior.

Quantitative Assessment of the Impact on Large Early Release Frequency (LERF)

Calvert Cliffs Nuclear Power Plant (CCNPP) received a similar Request for Additional Information (RAI) to address "how the potential leakage due to age-related degradation mechanisms were factored into the risk assessment for [their] Integrated Leakage Rate Test (ILRT) one-time extension." The CCNPP response to this RAI is documented in correspondence from Constellation Nuclear to the NRC, dated March 27, 2002 (TAC No. MB3929). In similar RAIs to other licensees, the NRC has previously indicated that the approach taken by CCNPP is an acceptable method to address this question. An approach similar to the CCNPP method was therefore used to respond to that question within the HBRSEP, Unit No. 2, letter dated June 19, 2002. A summary of the associated calculation is provided below, with additional detail regarding the basis for this approach having been provided in the CCNPP letter dated March 27, 2002, and in the HBRSEP, Unit No. 2, letter dated June 19, 2002.

The following summary updates the analysis presented in the June 19, 2002 letter to include two important changes: (1) Based on information in the September 17, 2003 RAIs regarding four containment degradation events, the historical liner flaw likelihood was recalculated, and (2) An additional year has passed since the CCNPP calculation method was developed; the additional year of exposure time was added to the 5.5 years assumed previously, and results were recalculated.

Step	Description	Containment Cylinder and Dome 83%		Basemat Region 17%	
1	Historical Liner Flaw Likelihood Failure Data: Containment location specific Success Data: Based on 70 steel-lined Containments and 6.5 years since the 10 CFR 50.55 requirement for periodic visual inspections of containment surfaces	Events: 4 4/(70*6.5) = 8.8E-3/year		Events: 0 Assume half a failure 0.5/(70*6.5) = 1.1E-3/year	
2	Age Adjusted Liner Flaw Likelihood	Year	Failure Rate	Year	Failure Rate
	During 15-year interval, assumed failure rate doubles every five years (14.9% increase per year). The average for 5^{th} to 10^{th} year was set to the historical failure rate. (See Calvert Cliffs response for basis)	1	3.47E-3	1	4.3E-4
		avg 5-10	8.8E-3	avg 5-10	1.1E-3
		15	2.4E-2	15	3.04E-3
		15 year avg	1.04E-2	15 year avg	1.39E-3
3	Increase in Flaw Likelihood between 3 and 15 years Uses age adjusted liner flaw likelihood (Step 2), assuming failure rate doubles every five years. (See Calvert Cliffs response for basis)	14%		1.87%	

4	Likelihood of Breach in Containment given Liner Flaw The upper end pressure is consistent with the H.B. Robinson Probabilistic Risk Assessment (PRA) Level 2 analysis. 0.1% is assumed for the lower end. Intermediate failure likelihoods are determined through interpolation. The basemat is assumed to be 1/10 of the cylinder/dome analysis.	Pressure (psia)	Likelihood of Breach	Pressure (psia)	Likelihood of Breach
		20	0.1%	20	0.01%
		57 (ILRT)	0.8%	57 (ILRT)	0.08%
		100	8.3%	100	0.83%
		120	25.2%	120	2.52%
		145	100%	145	10%
5	Visual Inspection Detection Failure Likelihood	77% The containment visual inspection failure likelihood was estimated to be 0.77. Approximately 74% of the containment cylinder and dome liner is covered by insulation and is not readily visible. It was assumed that the remaining approximately 26% which is visible would be visually examined during Refueling Outages. A 5% probability was then assumed that visible failures would not be detected, and a 5% probability was assumed that flaws could exist, but would not be visible. 0.74 + (0.26 * (0.05 + 0.05))		100% Cannot be inspected visually	
6	Likelihood of Non-Detected Containment Leakage	0.086% 14% * 0.8% * 77%		0.0015% 1.87% * 0.08% * 100%	

Therefore, the Total Likelihood of Non-Detected Containment Leakage = 0.086% + 0.0015% = 0.088%.

As discussed in the HBRSEP, Unit No. 2, letter dated June 19, 2002, the non-LERF core damage frequency (CDF) due to internal events, with sequences excluded that cannot contribute to LERF, is 2.21E-5 per year. If all non-detectable containment leakage events are considered to be LERF, then the increase in LERF associated with containment liner corrosion, based on going from an ILRT frequency of three times per 10 years to once per 15 years, can be expressed as:

0.088% x 2.21E-5 per year = 1.9E-8 per year

which is considered a very small contribution.