H. L. Sumner, Jr.
Vice President
Hatch Project

Southern Nuclear Operating Company, Inc.
Post Office Box 1295
Birmingham, Alabama 35201

Tel 205.992.7279

SOUTHERN COMPANY
Energy to Serve Your World℠

September 7, 2004

NL-04-1647

Docket No.: 50-366

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555-0001

Edwin I. Hatch Nuclear Plant – Unit 2
Technical Specification Revision Request
Integrated Leakage Rate Testing Interval Extension

Ladies and Gentlemen:

By letter dated April 26, 2004, Southern Nuclear Operating Company (SNC) submitted a request to revise the Edwin I. Hatch Nuclear Plant Unit 2 Technical Specifications. The proposed change was to revise TS section 5.5.12, ("Primary Containment Leakage Rate Testing Program") to reflect a one-time deferral of the Type A Containment Integrated Leak Rate Test (ILRT). By letter dated August 17, 2004, SNC provided additional information and indicated that an additional request for information concerning risk assessment would be addressed in separate correspondence. By this letter, SNC provides the additional information concerning risk assessment requested from the NRC staff on July 6, 2004. The questions and responses are provided in the Attachment. This submittal completes the SNC response to the requests for additional information for the subject Technical Specification revision request.

Mr. H. L. Sumner, Jr. states he is a Vice President of Southern Nuclear Operating Company, is authorized to execute this oath on behalf of Southern Nuclear Operating Company and to the best of his knowledge and belief, the facts set forth in this letter are true.

This letter contains no NRC commitments. If you have any questions, please advise.

Respectfully submitted,

SOUTHERN NUCLEAR OPERATING COMPANY

Davis Sumner

H. L. Sumner, Jr.

Sworn to and subscribed before me this 7 *day of* September, *2004.*

Valerie A O'Brien
Notary Public

My commission expires: 4/28/07

A017

HLS/IL/daj

Attachment: Response to RAI for the ILRT Extension Risk Assessment

cc: Southern Nuclear Operating Company
Mr. J. T. Gasser, Executive Vice President
Mr. G. R. Frederick, General Manager – Plant Hatch
RType: CHA02.004

U. S. Nuclear Regulatory Commission
Dr. W. D. Travers, Regional Administrator
Mr. C. Gratton, NRR Project Manager – Hatch
Mr. D. S. Simpkins, Senior Resident Inspector – Hatch

State of Georgia
Mr. L. C. Barrett, Commissioner – Department of Natural Resources

Attachment

Hatch Nuclear Plant Unit 2

RESPONSE TO RAI FOR THE ILRT EXTENSION RISK ASSESSMENT

RAI #1: The risk assessment methodology used to support the ILRT interval extension for Hatch is based on a methodology developed by EPRI in 1994. A revision to this method was developed for NEI by EPRI in 2001, and corrected and improved the original method in several areas. Based on an Nuclear Regulatory Commission (NRC) staff assessment, the revised method would indicate larger risk impacts (e.g., ΔLERF) for the ILRT interval extension than the original. The revised method (termed the NEI Interim Guidance) was used to support a similar ILRT extension request for another SNC plant (Farley) but was not used for the Hatch application. In view of the non-conservative nature of the original EPRI methodology, please provide a re-assessment of the risk impacts of the requested change for Hatch Nuclear Plant Unit 2 based on the NEI Interim Guidance.

RESPONSE #1: A previous analysis [1] was performed to evaluate the risk impact of extending the Integrated Leak Rate Test (ILRT) interval for Hatch Nuclear Plant Unit 2. The Unit 2 analysis was based on the previously submitted and approved analysis for Unit 1. As such, the methodology used in that analysis was performed prior to the release of the approach developed by NEI [2] for performing assessments of one-time extensions for Containment ILRT surveillance intervals. It is noted that the approach used for Hatch Unit 2 in the submittal is very similar to the approach used in the NEI Guidance so that the presented results in the original submittal are amenable to the format required in the NEI Interim guidance. The analysis below provides a re-assessment of the requested change for Hatch Nuclear Plant Unit 2 based on the NEI Interim Guidance.

The method chosen to display the results is according to the eight (8) accident classes consistent with these two reports. Table 1-1 lists these accident classes.

The steps taken to perform this risk assessment evaluation are then as follows:

Step A- Quantify the base-line risk in terms of frequency per reactor year for each of the eight accident classes presented in Table 1-1. (This encompasses Step 1 from the NEI Interim Guidance.)

Step B - Develop plant-specific person-rem dose (population dose) for each of the eight accident classes, and multiply the frequency from Step A by the population dose. (This encompasses Steps 2, 3, and 4 from the NEI Interim Guidance.)

Step C - Evaluate risk impact of extending Type A test interval from 3 in 10 to 1 in 10 and 1 in 15 years. Also include the percent

of population dose rate affected by the ILRT frequency. (This encompasses Steps 5, 6, and 7 from the NEI Interim Guidance.)

Step D - Determine the change in risk in terms of Large Early Release Frequency (LERF) in accordance with RG 1.174 [4]. (This encompasses Step 8 from the NEI Interim Guidance.)

Step E - Determine the impact on the Conditional Containment Failure Probability (CCFP). (This encompasses Step 9 from the NEI Interim Guidance.)

Table 1-1
ACCIDENT CLASSES

Accident Classes (Containment Release Type)	Description
1	No Containment Failure
2	Large Isolation Failures (Failure to Close)
3a	Small Isolation Failures (Liner Breach)
3b	Large Isolation Failures (Liner Breach)
4	Small Isolation Failures (Failure to seal –Type B)
5	Small Isolation Failures (Failure to seal—Type C)
6	Other Isolation Failures (e.g., Dependent Failures)
7	Failures Induced by Phenomena (Early and Late) – refined to Class 7a (Early) and 7b (Late) for the Hatch assessment
8	Bypass (Interfacing System LOCA)

STEP A - QUANTIFY THE BASE-LINE RISK IN TERMS OF FREQUENCY PER REACTOR YEAR

As described in the NEI guidance document [2], the extension of the Type A interval does not influence those accident progressions that already involve large containment isolation failures, Type B or Type C testing, or containment failure induced by severe accident phenomena. As such, only Class 1, Class 3a, and Class 3b are impacted by the ILRT interval.

Class 1 Sequences. This group consists of all core damage accident progression bins for which the containment remains intact (modeled as Technical Specification Leakage). The frequency per year is determined by subtracting all containment failure end states from the total CDF. This is consistent with the approach used in the original assessment [1], but results in a slightly different number since the contributions from Release Categories 3a and 3b are different when using the NEI guidance. This ends up being equal to 9.80E-6/yr in the base case analysis for this revised assessment.

Class 2 Sequences. This group consists of all core damage accident progression bins for which a failure to isolate the containment occurs. The frequency per year for these sequences is 5.5E-9/yr as indicated in Table 5-4 in Attachment 1 of the submittal. Additionally, what was classified as Release Category 6 in the original submittal (i.e., 5.0E-9/yr) is included in Class 2 for consistency with the NEI guidance methodology. This change does not impact the analysis since the associated person-rem is the same for both contributions. As such, the total Class 2 frequency for this re-assessment is 1.05E-8/yr.

Class 3 Sequences. This group consists of all core damage accident progression bins for which a pre-existing leak in the containment structure (e.g., containment liner) exists. The containment leak for these sequences can be either small ($2L_a$ to $35L_a$) or large ($>35L_a$).

The respective frequencies per year are determined from the NEI guidance as follows:

$PROB_{class_3a}$ = probability of small pre-existing containment liner leak
= 0.027

$PROB_{class_3b}$ = probability of large pre-existing containment liner leak
= 0.0027

Instead of applying these probabilities directly, further guidance was provided by NEI in a follow-on letter [3] to their initial ILRT guidance document. NEI issued additional information concerning the potential that the calculated delta LERF values for several plants may fall above the "very small change" guidelines of the NRC regulatory guide 1.174 [4]. This additional NEI information includes a discussion of conservatisms in the quantitative guidance for delta LERF. NEI describes ways to demonstrate that, using plant-specific calculations, the delta LERF is smaller than that calculated by the simplified method.

The supplemental information states:

> The methodology employed for determining LERF (Class 3b frequency) involves conservatively multiplying the CDF by the failure probability for this class (3b) of accident. This was done for simplicity and to maintain conservatism. However, some plant-specific accident classes leading to core damage are likely to include individual sequences that either may already (independently) cause a LERF or could never cause a LERF, and are thus not associated with a postulated large Type A containment leakage path (LERF). These contributors can be removed from Class 3b in the evaluation of LERF by multiplying the Class 3b probability by only that portion of CDF that may be impacted by type A leakage.

The application of this additional guidance for Hatch involves the following:

- The Class 2, Class 7a, and Class 8 sequences are subtracted from the CDF that is applied to Class 3b. To be consistent, the same change is made to the Class 3a CDF, even though these events are not considered LERF. Class 2, Class 7a, and Class 8 events refer to sequences with large pre-existing containment isolation failures, early phenomenological containment failures, or containment bypass events. These sequences are already considered to contribute to LERF in the Hatch Level 2 PSA analysis.

As such, the failure probabilities on those cases that are already LERF scenarios are subtracted in the derivation of the Class 3a and Class 3b base case frequency determination.

CLASS_3A_FREQUENCY = 0.027 * (CDF – Class 2 – Class 7a – Class 8)
= 0.027 * (1.24E-05 – 1.05E-08 – 2.0E-6 – 1.65E-7) = 2.76E-7/yr

CLASS_3B_FREQUENCY = 0.0027 * (CDF – Class 2 – Class 7a – Class 8)
= 0.0027 * (1.24E-05 – 1.05E-08 – 2.0E-6 – 1.65E-7) = 2.76E-8/yr

For this analysis, the associated containment leakage for Class 3a is $10L_a$ and for Class 3b is $35L_a$. These assignments are consistent with the NEI Interim Guidance.

<u>Class 4 Sequences</u>. This group consists of all core damage accident progression bins for which containment isolation failure-to-seal of Type B test components occurs.

Because these failures are detected by Type B tests which are unaffected by the Type A ILRT, this group is not evaluated any further in the analysis.

Class 5 Sequences. This group consists of all core damage accident progression bins for which a containment isolation failure-to-seal of Type C test components. Because the failures are detected by Type C tests which are unaffected by the Type A ILRT, this group is not evaluated any further in this analysis.

Class 6 Sequences. This group is similar to Class 2. These are sequences that involve core damage accident progression bins for which a failure-to-seal containment leakage due to failure to isolate the containment occurs. These sequences are dominated by misalignment of containment isolation valves following a test/maintenance evolution. As stated above, the previously assigned frequency to this Class has been included in the Class 2 representation.

Class 7 Sequences. This group consists of all core damage accident progression bins in which containment failure induced by severe accident phenomena occurs. For the Hatch analysis, this had been subdivided into two separate contributions (Early and Late). This distribution is maintained in this analysis consistent with the information in Table 5-4 in Attachment 1 of the Unit 2 submittal. The Class 7a frequency (Early Failures) is 2.0E-6/yr and the Class 7b frequency (Late Failures) is 1.1E-7/yr.

Class 8 Sequences. This group consists of all core damage accident progression bins in which containment bypass occurs. The frequency per year for these sequences is 1.65E-7/yr as indicated in Table 5-4 in Attachment 1 of the submittal.

Summary of Accident Class Frequencies

In summary, the accident sequence frequencies that can lead to radionuclide release to the public have been re-derived consistent with the definitions of accident classes defined in the NEI Interim Guidance. Table 1-2 summarizes these accident frequencies by accident class for Hatch.

Table 1-2

RADIONUCLIDE RELEASE FREQUENCIES AS A FUNCTION OF ACCIDENT CLASS (HATCH BASE CASE)

Accident Classes (Containment Release Type)	Description	NEI Methodology
1	No Containment Failure	9.80E-6
2	Large Isolation Failures (Failure to Close)	1.05E-8
3a	Small Isolation Failures (liner breach)	2.76E-7
3b	Large Isolation Failures (liner breach)	2.76E-8
4	Small Isolation Failures (Failure to seal –Type B)	NA
5	Small Isolation Failures (Failure to seal—Type C)	NA
6	Other Isolation Failures (e.g., dependent failures)	NA
7a	Failures Induced by Phenomena (Early)	2.01E-6 (1)
7b	Failures Induced by Phenomena (Late)	1.10E-7
8	Bypass (Interfacing System LOCA)	1.65E-7
CDF	All CET end states	1.24E-5

(1) Reported as 2.0E-6 in the original submittal. This value was expanded to 2.01E-6 for this assessment to maintain the appropriate initial LERF total.

STEP B - DEVELOP PLANT-SPECIFIC PERSON-REM DOSE (POPULATION DOSE) PER REACTOR YEAR

Plant-specific release analyses were performed to estimate the person-rem doses to the population within a 50-mile radius from the plant. The values developed for use in the original submittal are consistent with the NEI guidance. These values, when combined with the information in Table 1-2, yield the Hatch baseline mean consequence measures for each accident class. These results are presented in Table 1-3.

Table 1-3
HATCH UNIT 2
ANNUAL DOSE (PERSON-REM/YR) AS A FUNCTION OF ACCIDENT CLASS CHARACTERISTIC OF CONDITIONS FOR ILRT REQUIRED 3/10 YEARS (I.E., REPRESENTATIVE OF ILRT DATA)

Accident Classes (Containment Release Type)	Description	Frequency (per Rx-yr)	Person-Rem (50 miles)	Person-Rem/yr (50 miles)
1	No Containment Failure [1]	9.80E-6	1963	1.92E-2
2	Large Isolation Failures (Failure to Close)	1.05E-8	1.15E+6	1.21E-2
3a	Small Isolation Failures (liner breach)	2.76E-7	19,630	5.41E-3
3b	Large Isolation Failures (liner breach)	2.76E-8	68,705	1.89E-3
4	Small Isolation Failures (Failure to seal –Type B)	NA	NA	NA
5	Small Isolation Failures (Failure to seal—Type C)	NA	NA	NA
6	Other Isolation Failures (e.g., dependent failures)	NA	NA	NA
7a	Failures Induced by Phenomena (Early)	2.01E-6	1.06E+6	2.14
7b	Failures Induced by Phenomena (Late)	1.10E-7	5.70E+5	6.27E-2
8	Bypass (Interfacing System LOCA)	1.65E-7	1.15E+6	1.90E-1
CDF	**All CET End states (including very low and no release)**	**1.24E-5**		**2.43**

[1] Characterized as $1L_a$ release magnitude consistent with the derivation of the ILRT non-detection failure probability for ILRTs. Release Category 3a and 3b include failures of containment to meet the Technical Specification leak rate.

STEP C - EVALUATE RISK IMPACT OF EXTENDING TYPE A TEST INTERVAL FROM 10-TO-15 YEARS

The next step is to evaluate the risk impact of extending the test interval from its current ten-year value to a fifteen-year interval. To do this, an evaluation must first be made of the risk associated with the ten-year interval since the base case is assumed to apply to a 3-year interval (i.e., a simplified representation of the 3-in-10 year interval).

Risk Impact due to 10-year Test Interval

Per the NEI guidance, Type A tests impact only Class 3 sequences. For Class 3 sequences, the release magnitude is not impacted by the change in test interval, (a small or large breach remains the same, even though the probability of not detecting the breach increases). Thus, only the frequency of Class 3 sequences is impacted. Therefore, for Class 3 sequences, the risk contribution is changed based on the NEI guidance by a factor of 3.33 compared to the base case values. The results of the calculation for a 10-year interval are presented in Table 1-4.

Table 1-4
HATCH UNIT 2
ANNUAL DOSE (PERSON-REM/YR) AS A FUNCTION OF ACCIDENT CLASS CHARACTERISTIC OF CONDITIONS FOR ILRT REQUIRED 1/10 YEARS

Accident Classes (Containment Release Type)	Description	Frequency (per Rx-yr)	Person-Rem (50 miles)	Person-Rem/yr (50 miles)
1	No Containment Failure	9.09E-6	1963	1.78E-2
2	Large Isolation Failures (Failure to Close)	1.05E-8	1.15E+6	1.21E-2
3a	Small Isolation Failures (liner breach)	9.18E-7	19,630	1.80E-2
3b	Large Isolation Failures (liner breach)	9.18E-8	68,705	6.31E-3
4	Small Isolation Failures (Failure to seal –Type B)	NA	NA	NA
5	Small Isolation Failures (Failure to seal—Type C)	NA	NA	NA
6	Other Isolation Failures (e.g., dependent failures)	NA	NA	NA
7a	Failures Induced by Phenomena (Early)	2.01E-6	1.06E+6	2.14
7b	Failures Induced by Phenomena (Late)	1.10E-7	5.70E+5	6.27E-2
8	Bypass (Interfacing System LOCA)	1.65E-7	1.15E+6	1.90E-1
CDF	**All CET End states (including very low and no release)**	**1.24E-5**		**2.44**

Risk Impact Due to 15-Year Test Interval

The risk contribution for a 15-year interval is calculated in a manner similar to the 10-year interval. The difference is in the increase in probability of leakage in Classes 3a and 3b. For this case, the value used in the analysis is a factor of 5.0 compared to the 3-year interval value as described in the NEI guidance. The results for this calculation are presented in Table 1-5.

Table 1-5
HATCH UNIT 2
ANNUAL DOSE (PERSON-REM/YR) AS A FUNCTION OF ACCIDENT CLASS CHARACTERISTIC OF CONDITIONS FOR ILRT REQUIRED 1/15 YEARS

Accident Classes (Containment Release Type)	Description	Frequency (per Rx-yr)	Person-Rem (50 miles)	Person-Rem/yr (50 miles)
1	No Containment Failure	8.58E-6	1963	1.69E-2
2	Large Isolation Failures (Failure to Close)	1.05E-8	1.15E+6	1.21E-2
3a	Small Isolation Failures (liner breach)	1.38E-6	19,630	2.71E-2
3b	Large Isolation Failures (liner breach)	1.38E-7	68,705	9.47E-3
4	Small Isolation Failures (Failure to seal –Type B)	NA	NA	NA
5	Small Isolation Failures (Failure to seal—Type C)	NA	NA	NA
6	Other Isolation Failures (e.g., dependent failures)	NA	NA	NA
7a	Failures Induced by Phenomena (Early)	2.01E-6	1.06E+6	2.14
7b	Failures Induced by Phenomena (Late)	1.10E-7	5.70E+5	6.27E-2
8	Bypass (Interfacing System LOCA)	1.65E-7	1.15E+6	1.90E-1
CDF	**All CET End states (including very low and no release)**	**1.24E-5**		**2.45**

The percent of the population dose rate affected by the ILRT frequency is determined as dictated in the NEI guidance based on the information provided in Table 1-3, Table 1-4, and Table 1-5 respectively, for each of the intervals.

$\%Dose_{ILRT}$ = (Dose from 3a and 3b) / Total Dose * 100%

$\%Dose_3$ = (5.41E-3 + 1.89E-3) / 2.43 * 100% = 0.3%

$\%Dose_{10}$ = (1.80E-2 + 6.31E-3) / 2.44 * 100% = 1.0%

$\%Dose_{15}$ = (2.71E-2 + 9.47E-3) / 2.45 * 100% = 1.5%

In summary, at the base interval, the percent dose contribution from the ILRT frequency is 0.3%, at the ten-year interval the contribution is 1.0%, and at a 15-year interval the contribution is 1.5%. This level of change is judged to be insignificant.

STEP D - DETERMINE THE CHANGE IN RISK IN TERMS OF LARGE EARLY RELEASE FREQUENCY (LERF)

Reg. Guide 1.174 [4] provides guidance for determining the risk impact of plant-specific changes to the licensing basis. Reg. Guide 1.174 defines very small changes in risk as resulting in increases of core damage frequency (CDF) below 10^{-6}/yr and increases in LERF below 10^{-7}/yr, and small changes in LERF as below 10^{-6}/yr. Because the ILRT does not impact CDF, the relevant metric is LERF.

The risk increase associated with extending the ILRT interval involves the potential that a core damage event that normally would result in only a small radioactive release from an intact containment could in fact result in a larger release due to the increase in probability of failure to detect a pre-existing leak. With strict adherence to the NEI guidance, 100% of the Class 3b contribution would be considered LERF. For Hatch, however, the Class 3b radionuclide release person-rem is significantly less than a typical LERF contributor as can be seen by comparing the relative population dose for Class 3b to that of Class 2 (6.87E+4 person-rem / 1.15E+6 person-rem) or 6%. This conservatism is noted to obtain a proper perspective when comparing the results to an absolute threshold value per Reg. Guide 1.174.

Additionally, it is worth noting that recent efforts by EPRI [5] were performed to provide better guidance when examining ILRT interval extension requests. This included the use of an expert elicitation process in determining the likelihood of Class 3a and Class 3b scenarios. To briefly summarize, the key insight from the expert elicitation process is that the previous NEI guidance was judged to be very conservative for evaluating the frequencies associated with the Class 3a and 3b scenarios. In fact, the expert elicitation process results in best estimate base case values for the Class 3a and Class 3b scenarios that are 3.88E-3 and 2.47E-4, respectively. These are both about an order of magnitude below the base case values utilized in the original NEI methodology

(i.e. 0.027 and 0.0027, respectively), and as such would result in an overall order of magnitude reduction in the results when compared to the results presented here.

In any event, 100% of the frequency of Class 3b sequences (consistent with the NEI Guidance methodology) can be used as a very conservative first-order estimate to approximate the potential increase in LERF from the ILRT interval extension. Based on a ten-year test interval from Table 1-4, the Class 3b frequency is 9.18E-08/yr; and, based on a fifteen-year test interval from Table 1-5, it is 1.38E-07/yr. Thus, the overall increase in LERF from extending the ILRT interval from 10 to 15 years is 4.6E-8/yr. Additionally, the change in LERF can also be measured from the baseline frequency. In this case, from Table 1-3, the Class 3b frequency is 2.76E-8/yr. Thus, the overall increase in LERF from extending the ILRT interval from 3 to 15 years is 1.1E-7/yr. As can be seen, even with a conservative characterization in determining the likelihood of Class 3b sequences, and with conservatively characterizing all of these releases as LERF (per the NEI methodology), the estimated change in LERF for Hatch is below the threshold criteria for a very small change when measured from the current interval, and only slightly above the threshold criteria when measured from the original interval.

STEP E - IMPACT ON THE CONDITIONAL CONTAINMENT FAILURE PROBABILITY (CCFP)

Another parameter that the NRC Guidance in Reg. Guide 1.174 states can provide input into the decision-making process is the consideration of change in the conditional containment failure probability (CCFP). The change in CCFP is indicative of the effect of the ILRT on all radionuclide releases, not just LERF. The conditional containment failure probability (CCFP) can be calculated from the risk calculations performed in this analysis. One of the difficult aspects of this calculation is providing a definition of the "failed containment." In this assessment, the CCFP is defined such that containment failure includes all radionuclide release end states other than the intact state. The conditional part of the definition is conditional given a severe accident (i.e., core damage).

The change in CCFP can be calculated by using the method specified in NEI guidance.

CCFP= [1-(Class 1 frequency + Class 3a frequency) / CDF] * 100%

$CCFP_3$ = [1-(9.80E-6 + 2.76E-7) / 1.24E-5] * 100% = 18.8%

$CCFP_{10}$ = [1-(9.09E-6 + 9.18E-7) / 1.24E-5] * 100% = 19.3%

$CCFP_{15}$ = [1-(8.58E-6 + 1.38E-6) / 1.24E-5] * 100% = 19.7%

This change in CCFP of less than 1% when measured from the original interval is judged to be insignificant.

RAI #2: The April 26, 2004, submittal provides risk impacts for a change in test frequency from 1 test in 10 years to 1 test in 15 years. Please provide the corresponding risk results (for ΔPerson-rem, ΔLarge Early Release Frequency (LERF), and ΔConditional Containment Failure Probability) for a change in test frequency from 3 tests in 10 years to 1 test in 15 years.

RESPONSE #2: The requested information is provided in Table 2-1 for both the information provided in the submittal as well as for the information presented in response to RAI #1 above. The information is derived from Tables 5-4, 5-5, and 5-6 from Attachment 1 of the original submittal. The information for the NEI methodology assessment results is derived from Tables 1-3, 1-4, and 1-5 in response to RAI #1 above. Note that the CCFP for the original submittal methodology has been recalculated here to be consistent with the NEI methodology approach for that same parameter.

Table 2-1
Summary of Risk Results for the Hatch Unit 2 ILRT Interval Extension Evaluation [1]

	Original Hatch Submittal		NEI Methodology for Hatch	
Figure of Merit	**10 to 15 year ILRT interval extension**	**3 to 15 year ILRT interval extension**	**10 to 15 year ILRT interval extension**	**3 to 15 year ILRT interval extension**
ΔPerson-rem /yr	2.441 @15 yrs – 2.439 @ 10 yrs = **1.7E-3**	2.441 @15 yrs – 2.436 @ 3 yrs = **4.7E-3**	2.45 @15 yrs – 2.44 @ 10 yrs = **1.1E-2**	2.45 @15 yrs – 2.43 @ 3 yrs = **2.7E-2**
ΔLERF / yr	3.00E-7 @15 yrs – 2.86E-7 @10 yrs = **1.4E-8**	3.00E-7 @15 yrs – 2.60E-7 @ 3 yrs = **4.0E-8**	1.38E-7 @15 yrs – 9.18E-8 @10 yrs = **4.6E-8**	1.38E-7 @15 yrs – 2.76E-8 @ 3 yrs = **1.1E-7**
ΔCCFP	20.82% @15 yrs – 20.65% @10 yrs = **0.17%**	20.82% @15 yrs – 20.57% @3 yrs = **0.25%**	19.66% @15 yrs – 19.29% @10 yrs = **0.37%**	19.66% @15 yrs – 18.77% @3 yrs = **0.89%**

[1] Note that due to round-off, the results do not always exactly match the arithmetic equivalents of the displayed values.

As can be seen above, the results are slightly higher using the NEI methodology compared to the methodology used in the submittal, but given the conservatisms associated with characterizing all of the increase in Class 3b as LERF for Hatch as was noted in the RAI #1 response, this small difference should not alter the conclusions from the original analysis.

RAI #3: In Enclosure 1 to the April 26, 2004, submittal, Section e, "Typical Questions", Item 5, it is stated that the attached risk assessment for Hatch provides: (1) a sensitivity evaluation considering potential corrosion impacts within the framework of the ILRT interval extension risk assessment, (2) a series of parametric sensitivity studies regarding the potential age-related corrosion effects on the steel liner, and (3) a discussion on the effects the ILRT extension would have on the total LERF (internal and external events) for Hatch. However, this information was not included in the attached risk assessment. Please provide the noted information.

RESPONSE #3 (Part 1): The analysis in Response to RAI #1 above was performed to evaluate the risk impact of extending the Integrated Leak Rate Test (ILRT) interval for the Hatch Nuclear Plant Unit 2. That analysis was performed using the recommended approach developed by NEI [2, 3] for performing assessments of one-time extensions for containment ILRT surveillance intervals. The results of that analysis are summarized in Table 3-1. Based on the NEI guidance, only Classes 1, 3a, and 3b are impacted by the ILRT frequency. The results for these classes are highlighted in italics in the table. Supporting information regarding the change in dose, LERF, and CCFP is also provided on the bottom portion of the table for convenience in interpreting the results. This information is also redundant to what is presented in the responses to RAI #1 and RAI #2, but is provided here as a framework for presenting the potential corrosion impacts within the framework of the ILRT interval extension risk assessment.

Table 3-1
Hatch ILRT Cases: Base, 3 to 10, and 3 to 15 Yr Extensions Based on the NEI Methodology [1]

EPRI Class	Base Case 3 Years			Extend to 10 Years (Current Hatch Requirement)			Extend to 15 Years		
	CDF/Yr	Per-Rem	Per-Rem/Yr	CDF/Yr	Per-Rem	Per-Rem/Yr	CDF/Yr	Per-Rem	Per-Rem/Yr
1	*9.80E-6*	*1.96E+3*	*1.92E-2*	*9.09E-6*	*1.96E+3*	*1.78E-2*	*8.58E-6*	*1963*	*1.69E-2*
2	1.05E-8	1.15E+6	1.21E-2	1.05E-8	1.15E+6	1.21E-2	1.05E-8	1.15E+6	1.21E-2
3a	*2.76E-7*	*1.96E+4*	*5.41E-3*	*9.18E-7*	*1.96E+4*	*1.80E-2*	*1.38E-6*	*19,630*	*2.71E-2*
3b	*2.76E-8*	*6.87E+4*	*1.89E-3*	*9.18E-8*	*6.87E+4*	*6.31E-3*	*1.38E-7*	*68,705*	*9.47E-3*
4	NA	NA	NA	NA	NA	NA	NA	NA	NA
5	NA	NA	NA	NA	NA	NA	NA	NA	NA
6	NA	NA	NA	NA	NA	NA	NA	NA	NA
7a	2.01E-6	1.06E+6	2.14	2.01E-6	1.06E+6	2.14	2.01E-6	1.06E+6	2.14
7b	1.10E-7	5.70E+5	6.27E-2	1.10E-7	5.70E+5	6.27E-2	1.10E-7	5.70E+5	6.27E-2
8	1.65E-7	1.15E+6	1.90E-1	1.65E-7	1.15E+6	1.90E-1	1.65E-7	1.15E+6	1.90E-1
Total	**1.24E-5**		**2.43**	**1.24E-5**		**2.44**	**1.24E-5**		**2.45**
ILRT Dose Rate from 3a and 3b			7.31E-3			2.43E-2			3.65E-2
% of Total			0.30%			1.00%			1.49%
Total Delta Dose Rate (3 to 15 yr)									**2.68E-2**
LERF from 3b			2.76E-8			9.18E-8			1.38E-7
Delta LERF (3 to 15 yr)									**1.10E-7**
CCFP %			18.77%			19.29%			19.66%
Delta CCFP % (3 to 15 yr)									**0.89%**

[1] Note that due to round-off, the results do not always exactly match the arithmetic equivalents of the displayed values.

The increase in LERF due to extending the Hatch ILRT interval to 15 years from the original requirement is estimated to be 1.1E-7 /yr. As was noted in the RAI #1 response, this is judged to be a conservative characterization of the potential LERF increase for Hatch, and even as such is only slightly above the Regulatory Guide 1.174 acceptance criteria for "very small" changes in risk of 1.0E-7. Additionally, the dose increase was estimated to be 2.7E-2 Person-rem/yr, and the conditional containment

failure probability increase was estimated to be about 0.9%. Both of these increases are also considered to be "very small" although no official acceptance criteria exist for these parameters.

During follow-up RAIs on several other ILRT risk analyses, the NRC noted that inspections of some reinforced and steel containments (e.g., North Anna, Brunswick, D.C. Cook, Catawba/McGuire, Oyster Creek) have indicated degradation from the non-inspectable side of the liner/steel shell of primary containments. The major non-inspectable areas of the Mark I containment, such as Hatch, include the gap side of the drywell shell, and the portion of the shell below the drywell floor. Recent ILRT extension submittals, such as that submitted by Calvert Cliffs [6], have been expanded to include in the risk assessment an evaluation of the potential for age-related degradation from these non-inspectable areas. As such, the analysis below is a Hatch specific assessment of the potential for containment leakage due to age-related degradation in non-inspectable areas and the impact of this potential issue on the Hatch ILRT interval risk assessment results.

CONCEALED FLAW CORROSION ANALYSIS

The analysis utilizes a similar approach to that outlined in the Calvert Cliffs assessment [6] to estimate the likelihood and risk-implication of degradation-induced leakage occurring and going undetected in containment visual examinations during the extended test interval. It should be noted that the Calvert Cliffs analysis was performed for a concrete cylinder and dome containment with a steel liner whereas the Hatch containment is a BWR Mark I containment with a steel shell in the drywell region including the portion below the concrete drywell floor. As such, not all aspects of the Calvert Cliffs analysis are directly applicable to Hatch. Each of the analysis steps is described below with their relationship to the Calvert analysis noted where applicable.

The following approach is used to determine the change in likelihood, due to extending the ILRT, of detecting corrosion of the containment steel shell. This likelihood is then used to determine the resulting change in risk. Consistent with the Calvert Cliffs analysis, the following issues are addressed:

- Differences between the containment floor and other regions of containment;
- The historical steel shell flaw likelihood due to concealed corrosion;
- The impact of aging;
- The corrosion leakage dependency on containment pressure; and

- The likelihood that visual inspections will be effective at detecting a flaw.

Assumptions

A. The Oyster Creek incident is assumed to be applicable at Hatch for a concealed shell failure in the floor. In the Calvert Cliffs analysis, no applicable events were identified and 0.5 failures were assumed. Assuming 0.5 failures when 0 failures have occurred is a typical PRA approach. (See Table 3-2, Step 1.)

B. The two events used to estimate the liner flaw probability in the Calvert Cliffs analysis (i.e., North Anna 2 and Brunswick 2) are also assumed to be applicable to the BWR Mark I containment at Hatch, and the other events are judged to be not applicable. This is consistent with the Calvert approach. (See Table 3-2, Step 1.)

C. For consistency with the Calvert Cliffs analysis, the estimated historical flaw probability is calculated using a 5.5 year data period to reflect the years since September 1996 when 10 CFR 50.55a started requiring visual inspection. Additional success data was not used to limit the aging impact of this corrosion issue, even though inspections were being performed prior to this date (and have been performed since the time frame of the Calvert analysis), and there is no evidence that additional corrosion issues were identified. (See Table 3-2, Step 1.)

D. Consistent with the Calvert analysis, the corrosion-induced flaw likelihood is assumed to double every five years. This is based solely on judgment and is included in this analysis to address the increase in likelihood of corrosion as the steel shell ages. Sensitivity studies are included in the Part 2 response to this RAI that address doubling this rate every 10 years and every two years. (See Table 3-2, Steps 2 and 3.)

E. In the Calvert Cliffs analysis, the likelihood of the containment atmosphere reaching the outside atmosphere given a liner flaw exists was estimated as a function of the ILRT test pressure. This resulted in a failure probability of about 1.1% in the wall region for Calvert Cliffs. For Hatch, however, to bound the problem it is conservatively assumed that if an undetected flaw exists, that it will lead to containment failure 10% of the time during severe accidents as the containment pressurizes. (See Table 3-2, Step 4.) Consistent with the NEI guidance, however, this additional factor is not applied to those accident sequences that, regardless of the containment flaw issue, are already LERF (e.g., ISLOCA sequences). Sensitivity studies are included in the Part 2 response to this RAI that address the 10% failure probability assumption.

F. Consistent with the Calvert analysis, the likelihood of leakage escape (due to crack formation) in the floor region is considered to be less likely than the

containment wall region. (See Table 3-2, Step 4.) Sensitivity studies are included in the Part 2 response to this RAI that address this assumption.

G. Consistent with the Calvert analysis, a 5% visual inspection detection failure likelihood given the flaw is visible and a 5% likelihood of a non-detectable flaw is used. Therefore, a total undetected flaw probability of 10% is assumed in the base case analysis. (See Table 3-2, Step 5.) Sensitivity studies are included in Part 2 of this RAI response that evaluate total detection failure likelihood of 5% and 15%, respectively. Additionally, it should be noted that to date, all liner corrosion events have been detected through visual inspection and repaired.

H. An additional assumption that 90% of the liner flaws lead to EPRI release Class 3a, and 10% lead to EPRI release Class 3b was applied for Hatch. This is roughly consistent with the NEI Guidance methodology that shows a factor of 10 lower frequency on the Class 3b events compared to the Class 3a events. A sensitivity study is included in Part 2 of this RAI response that addresses a very conservative assumption that 100% of the flaws result in EPRI Class 3b scenarios.

Analysis

Table 3-2
Hatch Concealed Flaw Corrosion Analysis Steps

Step	Description	Containment Walls	Containment Floor
1	**Historical Steel Liner Flaw Likelihood**	Events: 2 (4 industry events, North Anna and Brunswick events assumed applicable to Hatch)	Events: 1 (1 industry event at Oyster Creek assumed applicable to Hatch)
	Failure Data: Containment location specific (applicable wall events and derived failure value is consistent with Calvert Cliffs analysis; one floor event assumed applicable for Hatch whereas Calvert assumed 0.5 failures).	2/(70 * 5.5) = **5.2E-3** (Based on 70 units with liners over 5.5 years)	1/(70 * 5.5) = **2.6E-3** (Based on 70 units with liners over 5.5 years)

Table 3-2
Hatch Concealed Flaw Corrosion Analysis Steps

Step	Description	Containment Walls		Containment Floor	
2	**Age Adjusted Steel Liner Flaw Likelihood** During 15-year interval, assume flaw likelihood doubles every five years (14.9% increase per year). The average for 5th to 10th year is set to the historical failure rate (consistent with Calvert Cliffs analysis).	Year	Flaw Likelihood	Year	Flaw Likelihood
		1	2.1E-3	1	1.0E-3
		avg 5-10	5.2E-3	avg 5-10	2.6E-3
		15	1.4E-2	15	7.0E-3
		15 year average = 6.27E-3		**15 year average = 3.14E-3**	
3	**Flaw Likelihood at 3, 10, and 15 years** Cumulative age adjusted liner flaw likelihood (Step 2), assuming failure rate doubles every five years (consistent with Calvert Cliffs analysis – See Table 6 of Reference [6]), for the 3 year, 10 year, and 15 year points in time.	**7.10E-3 (at 3 years)** **4.06E-2 (at 10 years)** **9.40E-2 (at 15 years)** (Note that the Calvert analysis presents the delta between 3 and 15 years of 8.7% to utilize in the estimation of the delta-LERF value. For this analysis, however, the values are calculated based on the 3, 10, and 15 year intervals consistent with the original evaluation shown in Table 3-1, and then the delta-LERF values are determined from there.)		**3.55E-3 (at 3 years)** **2.03E-2 (at 10 years)** **4.70E-2 (at 15 years)** (Note that the Calvert analysis assumed 0.5 failures and this analysis assumes 1 failure such that the values above represent twice the delta between 3 and 15 years to utilize in the estimation of the delta-LERF value.)	
4	**Likelihood of Breach in Containment Given Shell Flaw** Assume that a flaw in the wall leads to containment failure during the severe accident progression 10% of the time (compared to 1.1% in the Calvert Cliffs analysis). The floor failure probability is assumed to be 1% (compared to 0.11% in the Calvert analysis).	**10%**		**1%**	

Table 3-2
Hatch Concealed Flaw Corrosion Analysis Steps

Step	Description	Containment Walls	Containment Floor
5	**Visual Inspection Detection Failure Likelihood** Utilize assumptions consistent with Calvert Cliffs analysis.	**10%** 5% failure to identify visible flaws plus 5% likelihood that the flaw is not visible (not through-wall but could be detected by ILRT) All industry events have been detected through visual inspection, 5% visible failure detection is a conservative assumption.	**100%** Cannot be visually inspected.
6	**Likelihood of Non-Detected Corrosion-Induced Containment Leakage** (Steps 3 * 4* 5)	**7.10E-5 (at 3 years)** 7.10E-3 * 10% * 10% **4.06E-4 (at 10 years)** 4.06E-2 * 10% * 10% **9.40E-4 (at 15 years)** 9.40E-2 * 10% * 10%	**3.55E-5 (at 3 years)** 3.55E-3 * 1% * 100% **2.03E-4 (at 10 years)** 2.03E-2 * 1% * 100% **4.70E-4 (at 15 years)** 4.70E-2 * 1% * 100%

The total likelihood of the corrosion-induced, non-detected containment leakage is the sum of Step 6 for the containment walls and the containment floor as summarized below.

At 3 years: 7.10E-5 + 3.55E-5 = 1.07E-4 = 0.011%
At 10 years: 4.06E-4 + 2.03E-4 = 6.09E-4 = 0.061%
At 15 years: 9.40E-4 + 4.70E-4 = 1.41E-3 = 0.141%

Table 3-3 shows the results of the updated ILRT assessment including the potential impact from non-detected containment leakage scenarios assuming that 10% of the candidate sequences result in Class 3b (i.e., result in LERF) and the remainder result in Class 3a. Note that the impact of including the potential for corrosion-induced leakages compared to the Table 3-1 results are noted in parenthesis.

As discussed previously, the above calculated factors are applied to those core damage accidents that are not independently LERF. For example, the base case is calculated as follows:

- Per Table 3-1, the Class 3b frequency for the 3-year base case is 2.76E-8/yr.
- As discussed earlier in the RAI #1 response, the Hatch CDF associated with accidents that are not independently LERF is 1.24E-5/yr – 1.05E-8/yr - 2.0E-6/yr – 1.65E-7/yr (Class 2, Class 7a, and Class 8 are already LERF) = 1.02E-5/yr.
- The increase in the Base Case Class 3a and 3b frequency due to the concealed flaw corrosion issue is calculated as 1.02E-5/yr * 1.07E-4 = 1.09E-9/yr, where 1.07E-4 was previously shown to be the cumulative likelihood of non-detected containment leakage due to corrosion at 3 years.
- The Base Case Class 3b frequency including the concealed flaw corrosion issue is then calculated as 2.76E-8/yr + 0.10*1.09E-9/yr = 2.77E-8/yr, and the Class 3a frequency is 2.76E-7/yr + 0.90*1.09E-9/yr = 2.77E-7/yr.

Table 3-3
Final Results Including Concealed Flaw Corrosion Analysis
Hatch ILRT Cases: Base, 3 to 10, and 3 to 15 Yr Extensions [1]

	Base Case 3 Years			Extend to 10 Years			Extend to 15 Years		
EPRI Class	CDF/Yr	Per-Rem	Per-Rem/Yr	CDF/Yr	Per-Rem	Per-Rem/Yr	CDF/Yr	Per-Rem	Per-Rem/Yr
1	*9.80E-6*	*1.96E+3*	*1.92E-2*	*9.08E-6*	*1.96E+3*	*1.78E-2*	*8.57E-6*	*1.96E+3*	*1.68E-2*
2	1.05E-8	1.15E+6	1.21E-2	1.05E-8	1.15E+6	1.21E-2	1.05E-8	1.15E+6	1.21E-2
3a	*2.77E-7*	*1.96E+4*	*5.43E-3*	*9.24E-7*	*1.96E+4*	*1.81E-2*	*1.39E-6*	*1.96E+4*	*2.73E-2*
3b	*2.77E-8*	*6.87E+4*	*1.90E-3*	*9.24E-8*	*6.87E+4*	*6.35E-3*	*1.39E-7*	*6.87E+4*	*9.57E-3*
4	NA	NA	NA	NA	NA	NA	NA	NA	NA
5	NA	NA	NA	NA	NA	NA	NA	NA	NA
6	NA	NA	NA	NA	NA	NA	NA	NA	NA
7a	2.01E-6	1.06E+6	2.14	2.01E-6	1.06E+6	2.14	2.01E-6	1.06E+6	2.14
7b	1.10E-7	5.70E+5	6.27E-2	1.10E-7	5.70E+5	6.27E-2	1.10E-7	5.70E+5	6.27E-2
8	1.65E-7	1.15E+6	1.90E-1	1.65E-7	1.15E+6	1.90E-1	1.65E-7	1.15E+6	1.90E-1
Total	**1.24E-5**		**2.43**	**1.24E-5**		**2.44**	**1.24E-5**		**2.45**
ILRT Dose Rate from 3a and 3b			7.33E-3 (+2.0E-5)			2.45E-2 (+2.0E-4)			3.69E-2 (+4.0E-4)
% of Total			0.30% (+0.001%)			1.00% (+0.006%)			1.50% (+0.014%)
Total Delta Dose Rate (3 to 15 yr)									**2.71E-2 (+3.0E-4)**
LERF from 3b			2.77E-8 (+1.1E-10)			9.24E-8 (+6.2E-10)			1.39E-7 (+1.4E-9)
Delta LERF (3 to 15 yr)									**1.12E-7 (+1.3E-9)**
CCFP %			18.77% (+0.001%)			19.29% (+0.005%)			19.67% (+0.012%)
Delta CCFP % (3 to 15 yr)									**0.90% (+0.01%)**

[1] Note that due to round-off, the results do not always exactly match the arithmetic equivalents of the displayed values.

Based on the results in Table 3-3, it can be seen that including corrosion effects in the ILRT assessment would not significantly alter the results from the original analysis.

RESPONSE #3 (Part 2): Sensitivity cases were also developed to gain an understanding of the sensitivity of this analysis to the various key parameters. These results are summarized in Table 3-4. Again, the results are not significantly impacted unless the highly unlikely worst case assumptions are all applied together in the upper bound scenario.

Table 3-4
Concealed Flaw Corrosion Sensitivity Cases

Age (Step 2)	Containment Breach (Step 4)	Visual Inspection & Non-Visual Flaws (Step 5)	Likelihood Flaw is LERF (i.e., EPRI Class 3b)	LERF Increase From Corrosion (3 to 15 years)	Total LERF Increase From ILRT Extension (3 to 15 years)
Base Case Doubles every 5 yrs	**Base Case** (10% Wall, 1% Floor)	**Base Case** 10%	**Base Case** 10%	**Base Case** 1.33E-9	**Base Case** 1.12E-7
Doubles every 2 yrs	Base	Base	Base	3.04E-9	1.13E-7
Doubles every 10 yrs	Base	Base	Base	1.12E-9	1.11E-7
Base	*(100% Wall, 10% Floor)*	Base	Base	1.33E-8	1.24E-7
Base	*(1% Wall, 0.1% Floor)*	Base	Base	1.33E-10	1.10E-7
Base	Base	*15%*	Base	1.77E-9	1.12E-7
Base	Base	*5%*	Base	8.87E-10	1.11E-7
Base	Base	Base	*100%*	1.33E-8	1.24E-7
Base	Base	Base	*1%*	1.33E-10	1.10E-7
Lower Bound					
Doubles every 10 yrs	(10% Wall, 1% Floor)	5%	1%	7.48E-12	1.10E-7
Upper Bound					
Doubles every 2 yrs	(100% Wall, 10% Floor)	15%	100%	4.06E-7	5.16E-7

RESPONSE#3 (Part 3): A discussion on the effects the ILRT extension would have on the total LERF (internal and external events) for Hatch is provided in this portion of the response.

External hazards were evaluated in the Edwin Hatch Individual Plant Examination of External Events (IPEEE) Submittal in response to the NRC IPEEE Program (Generic Letter 88-20 Supplement 4). The IPEEE Program was a one-time review of external hazard risk to identify potential plant vulnerabilities and to understand severe accident risks. Hatch does not currently maintain external event PSA models and associated documentation. Although the external event hazards in the Hatch IPEEE were evaluated to varying levels of conservatism, the results of the Hatch IPEEE are nonetheless used in this RAI response to provide an assessment of the impact of external hazards on the conclusions of the Hatch Unit 2 ILRT interval extension risk assessment.

OVERVIEW OF HATCH IPEEE

HATCH IPEEE Internal Fires Analysis

The Hatch plant risk due to internal fires was evaluated in 1995 as part of the Hatch Individual Plant Examination of External Events (IPEEE) Submittal [7]. The EPRI Fire PRA Implementation Guide screening approaches and data were used to perform the Hatch IPEEE fire PRA study. The CDF contribution due to internal fires for Hatch Unit 2 was calculated at 5.4E-6/yr, and the LERF at 4.55-7/yr.

The IPEEE documentation for the fire induced core damage scenarios and the associated frequency results were reviewed in support of this assessment. The approximate breakdown of the Hatch Unit 2 IPEEE fire CDF risk profile is as shown in Table 3-5.

With respect to the Hatch Unit 2 fire IPEEE Level 2 PSA risk profile, a summary of the fire CDF as a function of subsequent containment release path is provided in Table 3-6.

Table 3-5
HATCH Unit 2 IPEEE Fire CDF as a Function of Accident Class

Accident Class	Description	% of Total Fire CDF
IA	Accident sequences involving loss of high-pressure coolant inventory makeup in which the reactor remains at high pressure	8%
IB	Accident sequences involving a loss of AC power (station blackout) and loss of coolant inventory makeup	44%
ID	Accident sequences involving loss of all coolant inventory makeup in which the reactor has been successfully depressurized	11%
II	Accident sequences involving loss of all containment heat removal, leading to containment failure and subsequent loss of coolant inventory makeup	30%
IIIB	Accident sequences initiated or resulting in a small or medium LOCA with inadequate high pressure coolant inventory makeup and the reactor is not depressurized	6%
IIIC	Accident sequences initiated or resulting in a medium or large LOCA for which the reactor is depressurized but low pressure coolant inventory makeup is inadequate	1%
IV	Accidents involving unmitigated failure to scram (ATWS)	negligible
V	Unisolated LOCA outside containment and inadequate coolant inventory makeup	negligible

Table 3-6
HATCH Unit 2 IPEEE Fire CDF as a Function of Radionuclide Release Pathway

Cntmnt Status	Description	% of Total Fire CDF
CN	Containment remains intact	14%
OT	Over-temperature failure of containment	54%
OP	Over-pressurization failure of containment	31%
VW	Wetwell venting initiated	1.4%
VD	Drywell venting initiated	negligible
CB	Containment bypass	negligible
CI	Containment isolation failure	0.6%

This information is used later in this response to provide quantitative insights into the impact of external hazard risk on the conclusions of this ILRT risk assessment.

HATCH IPEEE Seismic Analysis

The Hatch plant risk due to seismic events was also evaluated in 1995 as part of the Hatch Individual Plant Examination of External Events (IPEEE) Submittal [7]. Hatch performed a seismic margins assessment (SMA) following the guidance of NUREG-1407 and EPRI NP-6041. The SMA is a deterministic evaluation process that does not calculate risk on a probabilistic basis. No core damage frequency sequences were quantified as part of the IPEEE seismic risk evaluation.

Per NUREG-1407 and NRC Generic Letter 88-20, Supplement 4, the Hatch SMA was performed against a Seismic Margins Earthquake (SME) of 0.3g PGA (peak ground acceleration). Plant walkdowns and associated seismic capacity assessments were performed for the equipment on the Safe Shutdown Equipment List (SSEL). The Hatch Unit 2 IPEEE SSEL was defined in accordance with EPRI NP-6041, which includes defining two independent safe shutdown paths to be demonstrated as operable following the SME. The Hatch Unit 2 IPEEE seismic safe shutdown paths are:

- Primary: The primary path provides reactivity control (CRD), high pressure makeup (HPCI), RPV depressurization (ADS and SRVs), low pressure makeup (one loop of CS), and low pressure decay heat removal (one loop of RHR SPC).

- Alternate: The alternate path provides reactivity control (CRD), RPV depressurization (Manual initiation of SRVs), low pressure makeup (one loop of RHR LPCI), and low pressure decay heat removal (alternate shutdown cooling mode of RHR).

In accordance with NUREG-1407 and EPRI NP-6041, the seismic capacity assessments compare the estimated HCLPFs (high confidence low probability of failure) for the systems, structures, and components (SSCs) on the SSEL against the SME.

The Hatch Unit 2 IPEEE SMA analyses showed that all SSCs on the SSEL either already possessed a seismic HCLPF capacity of at least 0.3g PGA, or were modified to achieve the 0.3g PGA capacity.

The general conclusions of the Hatch IPEEE SMA analysis are as follows:

> *"The extensive evaluation of the design and location of Plant Hatch summarized in this report resulted in no fundamental weakness or vulnerability to seismic hazards. While no major plant changes were determined to be necessary, the seismic analysis identified modifications* [e.g., anchorage of DG relay panel, anchorage of HPCI room cooler ductwork, etc.] *of certain Unit 1 and Unit 2 components that were necessary to obtain a high-confidence-low-probability-of-failure capacity of at least 0.3 g peak ground acceleration. Modifications for* [all these items] *were completed in 1995."*

Although quantitative risk information is not directly available from the Hatch SMA IPEEE analysis, Reference [8] provides a simple method (called the Simplified Hybrid Method) for obtaining a seismic-induced CDF estimate based on results of an SMA analysis. Reference [8] has been cited by the NRC in other risk application guidance documents (e.g., Reference [9]) as a means of approximating seismic CDF for a licensee that has performed a seismic margins analysis. Reference [8] has shown that only the plant HCLPF seismic capacity is needed in order to estimate the seismic CDF within a precision of approximately a factor of two. The approach is as follows:

<u>Step 1:</u> Determine the plant HCLPF seismic capacity C_{HCLPF} from the SMA analysis

<u>Step 2:</u> Estimate the 10% conditional probability of failure capacity $C_{10\%}$ from:

$$C_{10\%} = F_\beta C_{HCLPF}$$

$$F_\beta = e^{1.044\beta}$$

where 1.044 is the difference between the 10% NEP standard normal variable (-1.282) and the 1% NEP standardized normal variable (-2.326).

Experience gained from high quality seismic PRA studies indicates that the plant damage state fragility determined by rigorous convolution will tend to have β_c values in the range of 0.30 to 0.35 (the plant damage state β_c value is equal to or less than the β_c values for the fragilities of the individual components that dominate the seismic risk). As such, the Simplified Hybrid method recommends:

$$C_{10\%} = 1.4C_{HCLPF}$$

Step 3: Determine hazard exceedance frequency $H_{10\%}$ that corresponds to $C_{10\%}$ from hazard curve.

Step 4: Determine seismic risk PF from:

$$PF = 0.5\ H_{10\%}$$

Using the Simplified Hybrid Method, an approximation of the Hatch seismic-induced CDF is performed here.

Step 1: If the SMA analysis assesses that the HCLPF of every SSC on the Seismic Safe Shutdown Paths is equal to or greater than the SME, the plant HCLPF is assessed as equal to the SME for the purposes of the Simplified Hybrid Method. Such is the case for the Hatch IPEEE SMA analysis. As the Hatch SME is 0.30g PGA (Peak Ground Acceleration), the Hatch plant HCLPF is 0.30g PGA for the purposes of this seismic CDF estimate.

Step 2: Using the relationship recommended above, the plant 10% capacity point (C10%) is estimated as 1.4 x 0.3g PGA = 0.42g PGA.

Step 3: The seismic hazard curve for the Hatch site, based upon EPRI NP-6395-D [10], is summarized in tabular form in Table 3-7. As can be seen from Table 3-7, the seismic hazard frequency associated with the 10% capacity point (0.42g PGA) is approximately 1.9E-6/yr.

Step 4: Using the relationship recommended above, the Hatch Unit 2 seismic-induced CDF is approximated as 0.50 x 1.9E-6/yr = 9.5E-7/yr.

Table 3-7
HATCH Site Seismic Hazard Curve
(EPRI Calculated Curve) [1]

Peak Ground Acceleration		EPRI Exceedance Frequency (1/yr, mean)
cm/s^2	g	
6	0.01	8.9E-3
60	0.06	3.0E-4
120	0.12	9.3E-5
226	0.23	1.4E-5
400	0.41	2.0E-6
560	0.57	6.7E-7
800	0.82	2.0E-7

[1] From Table 3-39 and Figure 3-115 of EPRI NP-6395-D, Appendix E [10].

The Simplified Hybrid Method only provides an overall seismic-induced CDF estimate and does not provide information as to the breakdown of seismic accident sequence types. A more rigorous analysis (e.g., a seismic PRA, or the Rigorous Hybrid Method referred to in References [8] and [11]) is required for such information. Such an analysis was not performed as part of this ILRT risk assessment. As such, the results of the NRC NUREG-1150 Peach Bottom seismic risk assessment [12] are used here to provide a reasonable approximation of the breakdown of seismic accident sequence types for the Hatch plant, they are as follows:

- Seismic-induced long-term LOOP/SBO loss of makeup ~50%
- Seismic-induced short-term LOOP/SBO loss of makeup ~30%
- Seismic-induced failure of major buildings or RPV (short-term loss of makeup) ~20%
- Seismic-induced ATWS scenarios << 1%
- Other seismic-induced accidents (e.g., non-LOOP/SBO, loss of DHR, etc.) << 1%

This information is used later in this response to provide quantitative insights into the impact of external hazard risk on the conclusions of this ILRT risk assessment.

Other External Hazards

In addition to internal fires and seismic events, the Hatch IPEEE Submittal analyzed a variety of other external hazards:

- High Winds/Tornadoes
- External Flooding
- Transportation and Nearby Facility Accidents
- Other External Hazards

The Hatch IPEEE analysis of high winds, tornadoes, external floods, transportation accidents, nearby facility accidents, and other external hazards was accomplished by reviewing the plant environs against regulatory requirements regarding these hazards. Based upon this review, it was concluded that Hatch meets the applicable Standard Review Plan requirements and therefore has an acceptably low risk with respect to these hazards. As such, these hazards were determined in the Hatch IPEEE to be negligible contributors to overall plant risk.

Accordingly, these other external event hazards are not included explicitly in this RAI response and are reasonably assumed not to impact the results or conclusions of the ILRT interval extension risk assessment.

IMPACT OF EXTERNAL HAZARD RISK ON ILRT RISK ASSESSMENT

The NEI Interim Guidance calculation of delta LERF performed for internal events is re-performed here including, in addition to internal event information, the Hatch IPEEE external event risk information.

As discussed previously, per the NEI Interim Guidance, the impact on the LERF risk measure due to the proposed ILRT interval extension is calculated as follows:

ΔLERF = (Frequency of EPRI Category 3b for 1-per-15 year ILRT interval) – (Frequency of EPRI Category 3b for 1-per-10 year ILRT interval)

Per the NEI Interim Guidance, the frequency per year for EPRI Category 3b is calculated as:

Frequency 3b = [3b conditional failure probability] x [CDF - (CDF with independent LERF + CDF that cannot cause LERF)]

The Hatch external event initiated CDF is approximately 5.4E-6/yr (internal fires) + 9.5E-7/yr (seismic) = 6.35E-6/yr. In addition, the following external event accident scenarios are excluded from the 3b frequency calculation because they cannot result in a LERF release (based on the timing of core damage) or independently result in LERF (regardless of ILRT postulated containment integrity issues):

- Fire-induced loss of decay heat removal scenarios (1.62E-6/yr)
  0.30 x 5.4E-6/yr = 1.62E-6/yr
- Fire-induced core damage with containment isolation failure (3.24E-8/yr)
  0.006 x 5.4E-6/yr = 3.24E-8/yr
- Seismic-induced long-term LOOP/SBO scenarios (4.75E-7/yr)
  0.50 x 9.5E-7/yr = 4.75E-7/yr
- Seismic-induced building or RPV failure scenarios (1.90E-7/yr)
  0.20 x 9.5E-7/yr = 1.90E-7/yr

Therefore, the baseline frequency of category 3b due to external events is calculated as (2.70E-03) x [(6.35E-6/yr) – (1.62E-6/yr + 3.24E-8/yr + 4.75E-7/yr + 1.90E-7/yr)] = 1.09E-8/yr.

Using the relationship described previously for the impact on 3b frequency due to increases in the ILRT surveillance interval (i.e., 3.33x increase for 10 yr, and 5.0x increase for 15 yr), the EPRI Category 3b frequency for the 1-per-10 year and 1-per-15 year ILRT intervals are calculated as 3.63E-8/yr and 5.45E-8/yr, respectively. Therefore, the change in the LERF risk measure due to extending the ILRT from 1-per-10 years to 1-per-15 years, including both internal and external hazard risk, is estimated as:

	3b Frequency (1-per-10 year ILRT)	3b Frequency (1-per-15 year ILRT)	LERF Increase
External Events Contribution	3.63E-8/yr	5.45E-8/yr	1.8E-8/yr
Internal Events Contribution	9.18E-8/yr	1.38E-7/yr	4.6E-8/yr
Combined (Internal + External)	1.28E-7/yr	1.92E-7/yr	6.4E-8/yr

Comparison to RG 1.174 Acceptance Guidelines

NRC Regulatory Guide 1.174, "An Approach for Using PRA in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis", provides NRC recommendations for using risk information in support of applications requesting changes to the license basis of the plant. As discussed previously, the risk acceptance criteria of RG 1.174 is used here to assess the ILRT interval extension.

The 6.4E-8/yr increase in LERF from extending the Hatch ILRT frequency from 1-per-10 years to 1-per-15 years falls into Region III ("Very Small Change" in risk) of the RG 1.174 acceptance guidelines, and is an acceptable plant change from a risk perspective.

Three sensitivity cases are discussed below:

- Case #1: 3-per-10 yr configuration used as reference point
- Case #2: LLNL seismic hazard curve used instead of EPRI curve
- Case #3: Combination of Case #1 and Case #2

Each of these sensitivity cases also show that the proposed ILRT frequency extension is acceptable from a risk perspective.

Sensitivity Case #1: This sensitivity case examines the impact on the results if the reference point is taken to be the 3-per-10 year configuration rather than the current 1-per-10 year configuration (as questioned in RAI #2). The delta LERF for this sensitivity case is calculated as:

	3b Frequency (3-per-10 year ILRT)	3b Frequency (1-per-15 year ILRT)	LERF Increase
External Events Contribution	1.09E-8/yr	5.45E-8/yr	4.4E-8/yr
Internal Events Contribution	2.76E-8/yr	1.38E-7/yr	1.1E-7/yr
Combined (Internal + External)	3.85E-8/yr	1.92E-7/yr	1.5E-7/yr

In this sensitivity case, the increase in LERF is calculated as 1.5E-7/yr and falls into Region II ("Small Change" in risk) of RG 1.174. Per RG 1.174, when the calculated increase in LERF due to the proposed plant change is in the range of 1E-7 to 1E-6 per reactor year, the risk assessment must also reasonably show that the total LERF is less than 1E-5/yr.

The Hatch Unit 2 LERF due to internal event accidents is 2.19E-6/yr [1]. The Hatch Unit 2 IPEEE LERF due to internal fire scenarios is 4.55E-7/yr. Explicit information on LERF due to seismic events is not available from the Hatch IPEEE; as such, this sensitivity case assumes 50% (a reasonably conservative estimate; note that for the internal events the percentage is 17%) of the estimated seismic CDF (9.50E-7/yr) results in LERF. Therefore, the total LERF for Hatch Unit 2 for this sensitivity case is calculated as 2.19E-6/yr + 4.55E-7/yr + (0.5 x 9.50E-7/yr) = 3.12E-6/yr, which is less than the RG 1.174 acceptance guideline of 1E-5/yr. Therefore, the results of this sensitivity case also indicate the proposed ILRT extension request is acceptable from a risk perspective.

Sensitivity Case #2: This sensitivity case examines the impact on the results if the Lawrence Livermore National Laboratory (LLNL) calculated seismic hazard curve for the Hatch plant is used instead of the EPRI hazard curve. The LLNL calculated seismic hazard curve for the Hatch plant is documented in Reference [13] and is summarized here in Table 3-8.

Table 3-8
HATCH Site Seismic Hazard Curve
(LLNL Calculated Curve)[1]

Peak Ground Acceleration		EPRI Exceedance Frequency (1/yr, mean)
cm/s^2	g	
50	0.05	6.13E-4
75	0.08	3.19E-4
150	0.15	9.66E-5
250	0.25	3.71E-5
300	0.31	2.58E-5
400	0.41	1.42E-5
500	0.51	8.64E-6
650	0.66	4.68E-6
800	0.82	2.80E-6
1000	1.02	1.57E-6

[1] From Appendix A of NUREG-1488 [13].

Using the LLNL curve and the Simplified Hybrid approach discussed previously, the Hatch Unit 2 seismic CDF is estimated in this sensitivity case at 6.8E-6/yr.

Revising the combined CDF and the seismic contributions, the baseline frequency of category 3b due to external events for this sensitivity case is calculated as (2.70E-03) x [(5.4E-6/yr + 6.8E-6/yr) – (1.62E-6/yr + 3.24E-8/yr + 0.5 * 6.8E-6/yr + 0.2 * 6.8E-6/yr)] = 1.56E-8/yr.

The combined (i.e., internal plus external initiators) delta LERF for this sensitivity case is calculated as:

	3b Frequency (1-per-10 year ILRT)	3b Frequency (1-per-15 year ILRT)	LERF Increase
External Events Contribution	5.20E-8/yr	7.80E-8/yr	2.6E-8/yr
Internal Events Contribution	9.18E-8/yr	1.38E-7/yr	4.6E-8/yr
Combined (Internal + External)	1.44E-7/yr	2.16E-7/yr	7.2E-8/yr

In this sensitivity case, the increase in LERF is calculated as 7.2E-8/yr and falls into Region III ("Very Small Change" in risk) of RG 1.174. Therefore, the results of this sensitivity case also indicate the proposed ILRT extension request is acceptable from a risk perspective.

Sensitivity Case #3: This sensitivity case examines the impact on the results if the assumptions of Case #1 (3-per-10 year configuration as reference point) and Case #2 (LLNL seismic curve) are both employed. The combined (i.e., internal plus external initiators) delta LERF for this sensitivity case is calculated as:

	3b Frequency (3-per-10 year ILRT)	3b Frequency (1-per-15 year ILRT)	LERF Increase
External Events Contribution	1.56E-8/yr	7.80E-8/yr	6.2E-8/yr
Internal Events Contribution	2.76E-8/yr	1.38E-7/yr	1.1E-7/yr
Combined (Internal + External)	4.32E-8/yr	2.16E-7/yr	1.7E-7/yr

In this sensitivity case, the increase in LERF is calculated as 1.7E-7/yr and falls into Region II ("Small Change" in risk) of RG 1.174. Per RG 1.174, when the calculated increase in LERF due to the proposed plant change is in the range of 1E-7 to 1E-6 per reactor year, the risk assessment must also reasonably show that the total LERF is less than 1E-5/yr.

Using the LERF information discussed previously for Case #1, the total LERF for Hatch Unit 2 for this sensitivity case is calculated as 2.19E-6/yr + 4.55E-7/yr + (0.5 x 6.80E-6/yr) = 6.05E-6/yr, which is less than the RG 1.174 acceptance guideline of 1E-5/yr. Therefore, the results of this sensitivity case also indicate the proposed ILRT extension request is acceptable from a risk perspective.

REFERENCES

[1] *Hatch Nuclear Plant Unit 2, Technical Specification Revision Request Integrated Leakage Rate Testing Interval Extension, Enclosure 1, Basis for Change Request and NRC Questions with SNC Response,* Submitted to NRC by Southern Nuclear, April 26, 2004.

[2] *Interim Guidance for Performing Risk Impact Assessments In Support of One-Time Extensions for Containment Integrated Leakage Rate Test Intervals,* Developed for NEI by John M. Gisclon, EPRI Consultant, William Parkinson and Ken Canavan, Data Systems and Solutions, November 2001.

[3] Anthony R. Pietrangelo, *One-time extensions of containment integrated leak rate test interval – additional information*, NEI letter to Administrative Points of Contact, November 30, 2001.

[4] *An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,* Regulatory Guide 1.174, July 1998.

[5] *Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals,* EPRI, Palo Alto, CA: 2003, 10009325.

[6] *Response to Request for Additional Information Concerning the License Amendment Request for a One-Time Integrated Leakage Rate Test Extension,* Letter from Mr. C. H. Cruse (Calvert Cliffs Nuclear Power Plant) to NRC Document Control Desk, March 27, 2002.

[7] *Individual Plant Examination for External Events, Edwin I. Hatch Nuclear Plant, Units 1 and 2*, Submitted to NRC by Southern Nuclear, 1995.

[8] Kennedy, R.P., *Overview of Methods for Seismic PRA and Margin Analysis Including Recent Innovations*, Proceedings of the OECD-NEA Workshop on Seismic Risk, Tokyo, Japan, August 1999. Available from OECD Nuclear Energy Agency, La Seine St.-Germain, 12 Boulevard des Iles, F-92130 Issy-les-Moulineaus, France.

[9] U.S. Nuclear Regulatory Commission, *Review Standard for Extended Power Uprates*, RS-001, Revision 0, December 2003.

[10] Electric Power Research Institute, *Probabilistic Seismic Hazard Evaluations at Nuclear Plant Sites in the Central and Eastern United States: Resolution of the Charleston Earthquake Issue*, NP-6395-D, April 1989.

[11] Campbell, R. and G. Hardy, *Applications Guide for Use of Seismic Margins Assessments in Quantitative Risk-Informed Decision-Making*, EPRI Report 1010998, July 2004.

[12] Lambright, J.A., Ravindra, M.K., et al., *Analysis of Core Damage Frequency: Peach Bottom, Unit 2 External Events*, NUREG/CR-4550, Vol. 4, Rev. 1, Part 3, December 1990.

[13] U.S. Nuclear Regulatory Commission, *Revised Livermore Seismic Hazard Estimates for Sixty-Nine Nuclear Power Plant Sites East of the Rocky Mountains*, NUREG-1488, Final Report, April 1994.