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RS-03-160

August 5, 2003

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

Quad Cities Nuclear Power Station, Units 1 and 2
Facility Operating License Nos. DPR-29 and DPR-30
NRC Docket Nos. 50-254 and 50-265

Subject: Additional Information Supporting the Request for Amendment to Technical Specification 5.5.12, "Primary Containment Leakage Rate Testing Program"

Reference: Letter from P. R. Simpson (Exelon Generation Company, LLC) to U. S. Nuclear Regulatory Commission, "Request for Amendment to Technical Specification 5.5.12, 'Primary Containment Leakage Rate Testing Program,'" dated February 27, 2003

In the referenced letter, Exelon Generation Company, LLC (EGC) requested an amendment to the facility operating licenses for Quad Cities Nuclear Power Station, Units 1 and 2. The proposed change revises Technical Specification (TS) 5.5.12, "Primary Containment Leakage Rate Testing Program," to reflect a one-time deferral of the primary containment Type A test to no later than July 22, 2009, for Unit 1, and no later than May 16, 2008, for Unit 2.

On June 10, 2003, the NRC requested additional information to complete its review of the license amendment request. The Attachment to this letter provides the requested information.

EGC has reviewed the information supporting a finding of no significant hazards consideration that was previously provided to the NRC in Attachment 2 of the referenced letter. The supplemental information provided in this submittal does not affect the bases for concluding that the proposed license amendment does not involve a significant hazards consideration.

If you have any questions or require additional information, please contact Mr. Kenneth M. Nicely at (630) 657-2803.

A017

I declare under penalty of perjury that the foregoing is true and correct.

Respectfully,

August 5, 2003
Executed on

Patrick R. Simpson
Patrick R. Simpson
Manager – Licensing
Mid-West Regional Operating Group

Attachment:
ERIN Report No. C467030403-5480, "Quad Cities Risk Assessment to Support NRC RAI Responses on Quad Cities ILRT," dated July 18, 2003

cc: Regional Administrator - NRC Region III
NRC Senior Resident Inspector - Quad Cities Nuclear Power Station
Office of Nuclear Facility Safety - Illinois Department of Nuclear Safety

ATTACHMENT

ERIN Report No. C467030403-5480, "Quad Cities Risk Assessment to Support NRC RAI Responses on Quad Cities ILRT," dated July 18, 2003

QUAD CITIES RISK ASSESSMENT TO SUPPORT NRC RAI RESPONSES ON QUAD CITIES ILRT

ERIN No. C467030403-5480

Prepared by: ______________________ Date: 07/18/2003

Reviewed by: ______________________ Date: 7/18/03

Approved by: ______________________ Date: July 18, 2003

Accepted by: N/A Date: N/A

Revisions:

Rev.	Description	Preparer/Date	Reviewer/Date	Approver/Date

ERIN Engineering and Research, Inc.
an SKF Group Company

RESPONSES TO NRC RAIs #1-#4 ON THE QUAD CITIES ILRT

RAI #1: Provide the technical justification for the assumption in the risk analysis that no long-term station blackout scenarios contribute to LERF. If this justification is based on timing arguments, provide a timeline for a representative scenario that includes consideration of the time at which the various emergency action levels are declared, the decision to evacuate is made, and the evacuation is initiated and completed. If this justification is based on source term magnitude, provide the estimated source terms for a representative scenario, and the definition of LERF used for this determination.

Response to RAI #1:

The Quad Cities long-term station blackout core damage accidents (Class IBL) result in non-LERF releases based on release timing and not on release magnitude (i.e., Quad Cities IBL core damage accidents have the potential to result in the entire spectrum of release magnitudes, including High magnitude releases; but, they can not result in Early releases). The following discussion focuses on the timing issues of Class IBL scenarios.

Typical of many industry PRAs, the Quad Cities PRA uses a radionuclide release categorization scheme comprised of two factors: release timing and release magnitude. Three timing categories are used, as follows:

1. Early (E) — Less than 5 hours
2. Intermediate (I) — Greater than or equal to 5 hours, but less than or equal to 24 hours
3. Late (L) — Greater than 24 hours.

The above accident release categories are based upon past experience concerning offsite accident response:

- 0-5 hours is conservatively assumed to include cases in which minimal offsite protective measures have been observed to be performed in non-nuclear accidents.
- 5-24 hours is a time frame in which much of the offsite nuclear plant protective measures can be assured to be accomplished.
- >24 hours are times at which the offsite measures can be assumed to be effective.

The timing categories are relative to the declaration of the Quad Cities General Emergency Action Level (per Exelon Nuclear's EP-AA-1006, Rev. 17, "Radiological Emergency Plan Annex For Quad Cities Generating Station").

The Quad Cities IBL accident scenarios include only those sequences in which high pressure injection (HPCI or RCIC) is available initially in the accident but subsequently fails. The representative IBL sequence for Quad Cities is sequence LOOP-20 of the LOOP event tree. Sequence LOOP-20 proceeds as follows:

Event	Time After Plant Trip
- Loss of Offsite Power initiating event	0
- Failure of emergency AC power (EDGs & SBODGs)	0
- HPCI/RCIC Initiation	~1 min.
- Battery depletion	4 hrs.
- Failure to blowdown (no DC power)	4 hrs.
- Loss of HPCI/RCIC (all) injection (no DC power)	4 hrs.
- RPV/containment parameters exceed HCTL curve	4 hrs.
- Time to core damage (1800F)	~5 hrs.
- Time to energetic containment failure (fastest, but low frequency, release scenario)	~8 hrs.

As can be seen from the above scenario, the Quad Cities IBL accident class results in a radionuclide release no earlier than approximately 8 hours after the LOOP initiator. The 8-hour release for the IBL core damage accident makes the conservative assumption that an early energetic containment failure mode (in-vessel steam explosion) occurs at about the time of core melt and relocation to the lower head (a low probability containment failure mode for the IBL accident).

The Quad Cities Emergency Plan (Recognition Category MG1) directs declaration of a General Emergency (i.e., the emergency classification with associated directives for evacuation) for the following station blackout conditions:

- Loss of power from TR-11 (TR-21) and TR-12 (TR-22)

  AND

- Failure of Emergency Diesel Generators and Station Blackout Diesels to supply power to ECCS buses 13-1 (23-1) and 14-1 (24-1)

  AND

- One of the following:
  - Restoration of power to bus 13-1 (23-1) or 14-1 (24-1) within 4 hours is NOT likely.

    OR

- Conditions are imminent that a Loss of Two Fission Product Barriers and Potential Loss of the Third (FG1) will occur prior to restoration of AC power to the Unit. (Imminent is defined as "mitigation actions have been ineffective and trended information indicates that the event or condition will occur within 2 hours.")

The loss of offsite and emergency power to buses 13-1 and 14-1 occurs at t=0 for sequence LOOP-20. The Quad Cities PRA assumes that the determination that AC power is not likely to be restored in the 4 hour time frame is made within the first hour into the accident. As such, a General Emergency is assumed declared at 1 hour into the event. The evacuation process would be initiated within minutes after the declaration and is estimated to be completed within 4 hours 10 minutes under worst assumed conditions based on site specific evacuation studies for weather and times of day variations (per Exelon Nuclear's EP-AA-1006, Rev. 17, "Radiological Emergency Plan Annex For Quad Cities Generating Station"). The earliest possible release for the IBL scenario occurs at approximately 8 hours (nearly 3 hours after evacuation is expected to be completed). Therefore, the IBL core damage accident is not an Early release.

RAI #2: Provide an assessment of the impact on risk results (Δperson-rem, ΔLERF, and ΔCCFP) if long-term station blackouts were not removed from the residual core damage frequency when determining the Category 3a and 3b frequencies.

Response to RAI #2:

The frequency of long term SBO core damage sequences is 3.14E-7/yr. Including long-term SBO scenarios in the EPRI Category 3a and 3b frequency calculations would not be typical or consistent with the NEI ILRT risk assessment methodology, but is performed here in response to this RAI. The results are shown in Table RAI #2-1.

The increase in LERF from the 1-in-10 year ILRT interval to the 1-in-15 year interval is determined to be 6.85E-9/yr when long-term SBO scenarios are included in the EPRI Category 3a and 3b frequencies. This represents an additional LERF increase of 1.42E-9/yr (a 26% increase) over the original ILRT submittal increase in LERF of 5.43E-9/yr. Including the long-term SBO contribution, however, still results in a LERF increase below the NRC Regulatory Guide 1.174 criterion of 1.0E-7/yr for "very small" risk change. The dose rate increase for the same extension interval with the SBO scenarios included is determined to be 1.53E-3 person-rem/yr, which represents an increase of 3.2E-4 person-rem/yr (a 26% increase) over the original ILRT submittal dose rate increase of 1.21E-3 person-rem/yr. The increase in the conditional containment failure probability (CCFP) is determined to be insignificant (0.3% with SBO sequences included versus 0.2% in the original submittal).

Table RAI #2-1

QUANTITATIVE RESULTS AS A FUNCTION OF ILRT INTERVAL

- Sensitivity Case for RAI #2, Include Long-Term SBO Contributions in Category 3a and 3b Frequencies -

EPRI Category	Dose (Person-Rem Within 50 miles)	Baseline (3-per-10 year ILRT)		Current (1-per-10 year ILRT)		Proposed (1-per-15 year ILRT)	
		Accident Frequency (per year)	Population Dose Rate (Person-Rem/Year Within 50 miles)	Accident Frequency (per year)	Population Dose Rate (Person-Rem/Year Within 50 miles)	Accident Frequency (per year)	Population Dose Rate (Person-Rem/Year Within 50 miles)
1	1.80E+03	4.52E-07	8.13E-04	3.46E-07	6.24E-04	2.71E-07	4.88E-04
2	5.88E+05	3.88E-09	2.28E-03	3.88E-09	2.28E-03	3.88E-09	2.28E-03
3a	1.80E+04	4.11E-08	7.39E-04	1.37E-07	2.46E-03	2.05E-07	3.70E-03
3b	6.30E+04	4.11E-09	2.59E-04	1.37E-08	8.62E-04	2.05E-08	1.29E-03
4	n/a	n/a	n/a	n/a	n/a	n/a	n/a
5	n/a	n/a	n/a	n/a	n/a	n/a	n/a
6	n/a	n/a	n/a	n/a	n/a	n/a	n/a
7	4.42E+05	1.58E-06	6.99E-01	1.58E-06	6.99E-01	1.58E-06	6.99E-01
8	5.88E+05	1.75E-08	1.03E-02	1.75E-08	1.03E-02	1.75E-08	1.03E-02
TOTALS:		2.10E-06	7.13E-01	2.10E-06	7.15E-01	2.10E-06	7.17E-01
Increase in Dose Rate (1)					2.14E-03		1.53E-03
Increase in LERF (2)				9.58E-09		6.85E-09	
Increase in $CCFP_{\%}$ (3)				0.4%		0.3%	

(1) The Increase in Dose Rate (person-rem/year) is with respect to the preceding ILRT interval, and is calculated by subtracting the Dose Rate totals.
(2) The Increase in LERF is with respect to the preceding ILRT interval, and is calculated by subtracting the EPRI Category 3b frequencies.
(3) The Increase in CCFP% (units in percentage points) is with respect to the preceding ILRT interval. The CCFP% is calculated as:

$$CCFP_{\%} = [1 - ((\text{Category 1 Frequency} + \text{Category 3a Frequency}) / CDF)] \times 100$$

Whether or not long term SBO scenarios are included in the EPRI Category 3a and 3b frequencies, the conclusion of the risk assessment does not change; that is, the Quad Cities ILRT interval extension to 1-in-15 year has a minimal impact on plant risk.

RAI #3: The offsite dose estimates are based on adjusted values for Peach Bottom rather than plant-specific values for Quad Cities. Provide an assessment of the impact on risk results if the doses were based on plant-specific values (e.g., the dose values provided in the Quad Cities Environmental Report for license renewal).

Response to RAI#3:

An assessment of the impact on risk results is requested to be based on Quad Cities specific dose values instead of the surrogate values used in the ILRT submittal. An ILRT sensitivity case is performed utilizing the Quad Cities dose values calculated as part of the Severe Accident Mitigation Alternatives (SAMA) evaluation (Table 4-4 of Appendix F of the License Renewal Application).

Table RAI #3-1 presents the results of the ILRT Sensitivity Case utilizing Quad Cities specific dose values (Column 2) for each EPRI Category. For comparison purposes, the dose values of the original ILRT submittal are utilized in the response to RAI #2 and are listed in Column 2 of Table RAI #2-1. The plant specific dose values for EPRI Categories 1, 3a, and 3b (the only categories affected by the ILRT extension per the NEI methodology) are nearly a factor of five less than those based on the NUREG-1150 Peach Bottom releases in the original ILRT submittal. It is noted that the accident frequency for each EPRI Category is independent of the dose and is therefore unchanged for this sensitivity case. The EPRI Category frequencies of the original ILRT submittal are based on Quad Cities specific data per the NEI methodology.

Using the Quad Cities specific dose values, the increase in LERF from the 1-in-10 year ILRT interval to the 1-in-15 year interval is determined to be 5.43E-9/yr which is the same as that calculated in the original ILRT submittal. (The LERF increase calculation is dependent only on the EPRI Category 3b frequencies which are unchanged for the sensitivity case). The increase in LERF remains below the NRC Regulatory Guide 1.174 criterion of 1.0E-7/yr for "very small" risk change. The dose rate increase for the same extension interval with Quad Cities specific dose values is 2.48E-4 person-rem/yr, which represents a decrease of 9.6E-4 person-rem/yr (a 79% decrease) from the original ILRT submittal interval dose rate increase of 1.21E-3 person-rem/yr. Similar to LERF, the increase in the conditional containment failure probability (CCFP) of 0.2% reported in the original ILRT submittal is unchanged for the sensitivity case. The change in CCFP calculation is dependent on the EPRI Category 1 and 3a frequencies which are unchanged for the sensitivity case.

Table RAI #3-1

QUANTITATIVE RESULTS AS A FUNCTION OF ILRT INTERVAL

- Sensitivity Case for RAI #3, Using Plant-Specific Dose Values -

EPRI Category	Plant-Specific Dose (Person-Rem Within 50 miles)	Baseline (3-per-10 year ILRT)		Current (1-per-10 year ILRT)		Proposed (1-per-15 year ILRT)	
		Accident Frequency (per year)	Population Dose Rate (Person-Rem/Year Within 50 miles)	Accident Frequency (per year)	Population Dose Rate (Person-Rem/Year Within 50 miles)	Accident Frequency (per year)	Population Dose Rate (Person-Rem/Year Within 50 miles)
1	3.68E+02(4)	4.61E-07	1.70E-04	3.78E-07	1.39E-04	3.18E-07	1.17E-04
2	1.97E+06(5)	3.88E-09	7.65E-03	3.88E-09	7.65E-03	3.88E-09	7.65E-03
3a	3.68E+03(6)	3.26E-08	1.20E-04	1.09E-07	4.00E-04	1.63E-07	6.00E-04
3b	1.29E+04(7)	3.26E-09	4.20E-05	1.09E-08	1.40E-04	1.63E-08	2.10E-04
4	n/a	n/a	n/a	n/a	n/a	n/a	n/a
5	n/a	n/a	n/a	n/a	n/a	n/a	n/a
6	n/a	n/a	n/a	n/a	n/a	n/a	n/a
7	8.54E+05(8)	1.58E-06	1.35E+00	1.58E-06	1.35E+00	1.58E-06	1.35E+00
8	3.76E+06(9)	1.75E-08	6.58E-02	1.75E-08	6.58E-02	1.75E-08	6.58E-02
TOTALS:		2.10E-06	1.42E+00	2.10E-06	1.42E+00	2.10E-06	1.42E+00
Increase in Dose Rate (1)					3.47E-04		2.48E-04
Increase in LERF (2)				7.60E-09		5.43E-09	
Increase in $CCFP_{\%}$ (3)				0.3%		0.2%	

(1) The increase in Dose Rate (person-rem/year) is with respect to the preceding ILRT interval, and is calculated by subtracting the Dose Rate totals.
(2) The increase in LERF is with respect to the preceding ILRT interval, and is calculated by subtracting the EPRI Category 3b frequencies.
(3) The increase in CCFP% (units in percentage points) is with respect to the preceding ILRT interval. The CCFP% is calculated as:

$$CCFP_{\%} = [1 - ((\text{Category 1 Frequency} + \text{Category 3a Frequency}) / \text{CDF})] \times 100$$

(4) Dose based on QC SAMA sequence L2-10 representing an intact containment.
(5) Dose based on QC SAMA sequence L2-1 representing the highest containment failure (non-containment bypass) dose.
(6) Dose based on 10 times the EPRI Category 1 dose, per NEI methodology.
(7) Dose based on 35 times the EPRI Category 1 dose, per NEI methodology.
(8) Dose based on a weighted average of QC SAMA sequences L2-1, L2-2, L2-4, L2-5, L2-7, and L2-8. The weighted average approach was utilized in the original ILRT submittal and is acceptable since the total frequency and dose associated with EPRI Category 7 does not change as part of the ILRT extension.
(9) Dose based on QC SAMA sequence L2-9 representing containment bypass.

Utilizing Quad Cities specific dose values, the conclusion of the risk assessment does not change; that is, the Quad Cities ILRT interval extension to 1-in-15 yr. has a minimal impact on plant risk.

RAI #4: Inspections of 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 containment. Please describe the uninspectable areas of the Quad Cities 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 to RAI #4:

As requested by this RAI, a separate risk assessment has been performed regarding the potential for containment leakage due to age-related degradation in non-inspectable areas and the impact of this potential issue on the Quad Cities ILRT interval risk assessment results. This analysis was performed using the same approach used by other industry plants (e.g., Calvert Cliffs) to respond to similar NRC RAIs. The results of this analysis are that the increase in LERF due to extending the ILRT interval from 1-in-10 years to 1-in-15 years is 6.43E-9/yr, of which 1.00E-9/yr is due to corrosion. This value is well below the threshold for "very small" changes in risk. Additionally, a series of parametric sensitivity studies regarding the potential age related corrosion effects on the steel liner indicate that even with very conservative assumptions, the conclusions from the original analysis would not change. Refer to Attachment 1 for the details of this analysis.

Attachment 1

IMPACT OF UNDETECTED STEEL LINER CORROSION ON THE QUAD CITIES ILRT EXTENSION RISK ASSESSMENT

IMPACT OF UNDETECTED STEEL LINER CORROSION ON THE QUAD CITIES ILRT EXTENSION RISK ASSESSMENT

Section 1
BACKGROUND

A previous analysis was performed to evaluate the risk impact of extending the Integrated Leak Rate Test (ILRT) interval for the Quad Cities Nuclear Generating Station [1]. That analysis was performed using the recommended approach developed by NEI for performing assessments of one-time extensions for containment ILRT surveillance intervals [2]. The results of that analysis are summarized in Table 1, which is a copy of Table 4-1 from Reference 1.

The risk increase from extending the ILRT interval from the original 3-in-10 year requirement to 1-in-15 years is quantified by the increase in LERF (the CDF is not impacted by the ILRT interval). The NRC Regulatory Guide 1.174 [3] defines very small changes in risk as resulting in increases in LERF below 1.0E-7/yr. The Regulatory Guide also states that when the calculated increase in LERF is in the range of 1.0E-6/yr to 1.0E-7/yr, applications will be considered only if it can be reasonably shown that the total LERF is less than 1.0E-5/yr. For Quad Cities the increase in LERF from the 3-in-10 year interval to the 1-in-15 year interval was determined to be 1.30E-8/yr, which is well below the very small change threshold.

As can also be seen in Table 1, the dose rate increase was determined to be 3E-3 person-rem/yr, which is only 0.4% above the 3-in-10 year value of 7.13E-1 person-rem/yr. The increase in the containment failure probability (CCFP) was determined to be 0.6%, which is also judged to be insignificant.

Table 1

QUANTITATIVE RESULTS AS A FUNCTION OF ILRT INTERVAL

EPRI Category	Dose (Person-Rem Within 50 miles)	Baseline (3-per-10 year ILRT)		Current (1-per-10 year ILRT)		Proposed (1-per-15 year ILRT)	
		Accident Frequency (per year)	Population Dose Rate (Person-Rem/Year Within 50 miles)	Accident Frequency (per year)	Population Dose Rate (Person-Rem/Year Within 50 miles)	Accident Frequency (per year)	Population Dose Rate (Person-Rem/Year Within 50 miles)
1	1.80E+3	4.61E-7	8.30E-4	3.78E-7	6.80E-4	3.18E-7	5.72E-4
2	5.88E+5	3.88E-9	2.28E-3	3.88E-9	2.28E-3	3.88E-9	2.28E-3
3a	1.80E+4	3.26E-8	5.87E-4	1.09E-7	1.96E-3	1.63E-7	2.93E-3
3b	6.30E+4	3.26E-9	2.05E-4	1.09E-8	6.84E-4	1.63E-8	1.03E-3
4	n/a	n/a	n/a	n/a	n/a	n/a	n/a
5	n/a	n/a	n/a	n/a	n/a	n/a	n/a
6	n/a	n/a	n/a	n/a	n/a	n/a	n/a
7	4.42E+5	1.58E-6	6.99E-1	1.58E-6	6.99E-1	1.58E-6	6.99E-1
8	5.88E+5	1.75E-8	1.03E-2	1.75E-8	1.03E-2	1.75E-8	1.03E-2
TOTALS:		2.10E-6(4)	7.13E-1	2.10E-6(4)	7.15E-1	2.10E-6(4)	7.16E-1
Increase in Dose Rate (1)					0.002		0.001
Increase in LERF (2)				7.60E-9		5.4E-9	
Increase in CCFP(%)(3)				0.3%		0.3%	

(1) The increase in dose rate (person-rem/year) is with respect to the results for the preceding ILRT interval, as presented in the table. For example, the increase in dose rate for the proposed 1-per-15 ILRT is calculated as: total dose rate for 1-per-15 year ILRT, minus total dose rate for 1-per-10 year ILRT. For each case, the dose rate increase is insignificant.

(2) The increase in Large Early Release Frequency (LERF) is with respect to the results for the preceding ILRT interval, as presented in the table. The change in LERF is determined by the change in the accident frequency of EPRI Category 3b. For example, the increase in LERF for the proposed 1-per-15 ILRT is calculated as: 3b frequency for 1-per-15 year ILRT, 1.63E-8/yr, minus 3b frequency for 1-per-10 year ILRT, 1.09E-8/yr, equals 5.4E-9/yr.

(3) The conditional containment failure probability (CCFP) is calculated as:

$$CCFP_{\%} = [1 - ((\text{Category \#1 Frequency} + \text{Category \#3a Frequency}) / \text{CDF})] \times 100\%$$

(4) Due to the NEI methodology and round-off, the total frequency of all severe accidents is slightly less than the QC reported CDF (approximately 4%).

Recently, the NRC issued Requests for Additional Information (RAIs) in response to the one-time relief request for the ILRT surveillance interval. The analysis that follows addresses the following RAI:

Request for Additional Information No. 4:

Inspections of 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 Quad Cities 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.

Section 2
STEEL LINER CORROSION ANALYSIS

The analysis utilizes the methods of the Calvert Cliffs liner corrosion analysis [4] to estimate the likelihood and risk-implications of degradation-induced leakage occurring undetected during the extended test interval. The Calvert Cliffs analysis was performed for a concrete cylinder and dome and a concrete basemat, each with a steel liner. The Quad Cities containment is a pressure-suppression BWR/Mark I type with a steel shell in the drywell region, including the portion below the concrete drywell floor. The shell is surrounded by a concrete shield.

The following approach is used to determine the change in likelihood of detecting corrosion of the steel containment shell, due to extending the ILRT,. 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 the containment
- The historical steel liner flaw likelihood due to concealed corrosion
- The impact of aging
- The corrosion leakage dependency on containment pressure
- The likelihood that visual inspections will be effective at detecting a flaw

ASSUMPTIONS

A. The Oyster Creek incident is assumed to be applicable for Quad Cities for a concealed shell in the floor. (See Table 2, Step 1.) In the Calvert Cliffs analysis, no applicable events were identified and 0.5 failures were assumed. For Quad Cities it will be assumed that there has been one failure, in industry experience, for the floor area.

B. The two corrosion events used to estimate the liner flaw probability in the Calvert Cliffs analysis are assumed to be applicable to the Quad Cities containment analysis. These events, one at North Anna Unit 2 and one at Brunswick Unit 2, were initiated from the non-visible (backside) portion of the containment liner.

C. For consistency with the Calvert Cliffs analysis, the estimated historical flaw probability is calculated using a 5.5 year data period. This reflects the span from September 1996 when 10 CFR 50.55a started requiring visual inspection and the time of the Calvert Cliffs analysis. Additional success data were 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 Cliffs analysis), and there is no evidence that additional corrosion issues were identified. (See Table 2, Step 1.)

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

E. In the Calvert Cliffs analysis, the likelihood of the containment atmosphere reaching the outside atmosphere given that a liner flaw exists was estimated (based on an assessment of the containment fragility curve versus the ILRT test pressure) as 1.1% for the containment walls and dome region and 0.11% (factor of ten less) for the basemat. For Quad Cities the containment failure probabilities are conservatively assumed to be 10% for the shell wall and 1% for the floor. Sensitivity studies are included that increase and decrease the probabilities by an order of magnitude. (See Table 2, Step 4.)

F. 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 2, Step 5.) Consistent with the Cavert Cliffs estimate of 85% of the interior wall surface being visible for inspection for the Calvert Cliffs analysis, Quad Cities estimates that at least 85% of the interior surface of the Quad Cities containment is inspectable. Sensitivity studies are included that use a total detection failure likelihood of 5% and 15%. Additionally, it should be noted that, to date, all liner corrosion events have been detected through visual inspection and repaired.

G. Consistent with the Calvert analysis, all non-detectable containment failures are assumed to result in early releases. This approach avoids a detailed analysis of containment failure timing and operator recovery actions.

ANALYSIS

Table 2

STEEL LINER CORROSION BASE CASE

<table>
<tr><th>Step</th><th>Description</th><th colspan="2">Containment Walls</th><th colspan="2">Containment Floor</th></tr>
<tr><td>1</td><td>Historical Steel Liner Flaw Likelihood

Failure Data: Containment location specific (applicable wall events and derived failure value is consistent with Calvert Cliffs analysis; one floor event assumed applicable for Quad Cities whereas the Calvert Cliffs analysis assumed 0.5 failures).</td><td colspan="2">Industry Applicable Events:
2
(North Anna and Brunswick events assumed to be applicable to Quad Cities)

2/(70 * 5.5) = 5.2E-3

(Based on 70 units with liners over 5.5 years)</td><td colspan="2">Industry Applicable Events:
1
(Oyster Creek event assumed applicable to Quad Cities)

1/(70 * 5.5) = 2.6E-3

(Based on 70 units with liners over 5.5 years)</td></tr>
<tr><td>2</td><td>Age Adjusted Steel Liner Flaw Likelihood

During 15-year interval, assume failure rate doubles at the end of every five years (which equates to a 14.9% increase per year). The average over the 5th through 10th year period is set equal to the historical failure rate of Step 1 (consistent with Calvert Cliffs analysis). These assumptions are used to calculate the flaw likelihood for each year (for a 15 year period)</td><td>Year
0
1
2
3
4
5
6
7
8
9
10
11
12
13
14
15</td><td>Flaw Likelihood
1.8E-03
2.1E-03
2.4E-03
2.7E-03
3.1E-03
3.6E-03
4.1E-03
4.7E-03
5.4E-03
6.2E-03
7.1E-03
8.2E-03
9.4E-03
1.1E-02
1.2E-02
1.4E-02</td><td>Year
0
1
2
3
4
5
6
7
8
9
10
11
12
13
14
15</td><td>Flaw Likelihood
8.9E-04
1.0E-03
1.2E-03
1.4E-03
1.6E-03
1.8E-03
2.1E-03
2.4E-03
2.7E-03
3.1E-03
3.6E-03
4.1E-03
4.7E-03
5.4E-03
6.2E-03
7.1E-03</td></tr>
<tr><td>3</td><td>Flaw Likelihood at 3, 10, and 15 years

This cumulative probability uses the age adjusted liner flaw likelihood of Step 2 (consistent with Calvert Cliffs analysis – See Table 6 of Reference [4]). For example, the 7.12E-03 (at 3 years) cumulative flaw likelihood is the sum of the year 1, year 2, and year 3 likelihoods of Step 2.</td><td colspan="2">7.12E-3 (at 3 years)
4.14E-2 (at 10 years)
9.66E-2 (at 15 years)

(Note that the Calvert Cliffs 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 the values are calculated based on the 3, 10, and 15 year intervals.)</td><td colspan="2">3.56E-3 (at 3 years)
2.07E-2 (at 10 years)
4.83E-2 (at 15 years)

(Note that the Calvert Cliffs analysis presents the delta between 3 and 15 years of 2.2% to utilize in the estimation of the delta-LERF value. For this analysis the values are calculated based on the 3, 10, and 15 year intervals.)</td></tr>
</table>

Table 2
STEEL LINER CORROSION BASE CASE

Step	Description	Containment Walls	Containment Floor
4	**Likelihood of Breach in Containment Given Steel Liner Flaw** The failure probability of the containment is assumed to be 10% (compared to 1.1% in the Calvert Cliffs analysis). The floor failure probability is assumed to be a factor of ten less, 1%, (compared to 0.11% in the Calvert Cliffs analysis).	**10%**	**1%**
5	**Visual Inspection Detection Failure Likelihood** Utilize assumptions consistent with Calvert Cliffs analysis.	**10%** 5% failure to identify visual flaws plus 5% likelihood that the flaw is not visible (not through-wall but could be detected by ILRT). All 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 Containment Leakage** (Steps 3 * 4 * 5)	**7.12E-5 (at 3 years)** 7.12E-3 * 10% * 10% **4.14E-4 (at 10 years)** 4.14E-2 * 10% * 10% **9.66E-4 (at 15 years)** 9.66E-2 * 10% * 10%	**3.56E-5 (at 3 years)** 3.56E-3 * 1% * 100% **2.07E-4 (at 10 years)** 2.07E-2 * 1% * 100% **4.83E-4 (at 15 years)** 4.83E-2 * 1% * 100%

Cumulative Likelihood of Non-Detected Containment Leakage Due to Corrosion

The total likelihood of the corrosion-induced, non-detected containment leakage is the sum in Step 6 for the containment walls and the containment floor:

At 3 years: 7.12E-5 + 3.56E-5 = 1.07E-4

At 10 years: 4.14E-4 + 2.07E-4 = 6.21E-4

At 15 years: 9.66E-4 + 4.83E-4 = 1.45E-3

Table 3 summarizes the results of the revised ILRT assessment including the potential impact from non-detected corrosion-induced containment leakage scenarios, with the assumption that all of these scenarios result in EPRI Class 3b (i.e., LERF). The impact of including the potential for corrosion-induced leakages compared to the original analysis [1] results are noted in parentheses.

The factors calculated above are applied to those core damage accidents that ARE NOT already independently LERF or that could never result in LERF. For example, the 3-in-10 year base case is calculated as follows:

- Per Table 1, the EPRI Class 3b frequency is 3.26E-9/yr.
- As discussed in Section 3.1 of Reference 1, the Quad Cities CDF associated with accidents that are not independently LERF or that could never result in LERF is 2.18E-6/yr – (3.14E-7/yr + 6.45E-7/yr + 1.75E-8/yr) = 1.20E-6/yr. (As explained in Reference 1, the total CDF of 2.10E-6/yr is 4% less than the value of 2.18E-6/yr used to calculate the initial Class 3b frequency due to the NEI methodology and round-off.)
- The increase in the base case Class 3b frequency due to the corrosion-induced concealed flaw issue is calculated as 1.20E-6/yr * 1.07E-4 = 1.29E-10/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 corrosion-induced concealed flaw issue is then calculated as 3.26E-9/yr + 1.29E-10/yr = 3.39E-9/yr.

Table 3

QUAD CITIES ILRT CASES: BASE, 1 IN 10, AND 1 IN 15 YR EXTENSIONS (Including Age Adjusted Steel Liner Corrosion Likelihood) (1)

EPRI Category	Dose (Per-Rem)	Base Case 3 In 10 Years		Extend to 1 In 10 Years		Extend to 1 In 15 Years	
		Core Damage Frequency (/yr)	Dose Rate (person-Rem/yr)	Core Damage Frequency (/yr)	Dose Rate (person-Rem/yr)	Core Damage Frequency (/yr)	Dose Rate (person-Rem/yr)
1	1.80E+3	4.61E-7	8.30E-4	3.77E-7	6.78E-4	3.16E-7	5.68E-4
2	5.88E+5	3.88E-9	2.28E-3	3.88E-9	2.28E-3	3.88E-9	2.28E-3
3a	1.80E+4	3.26E-8	5.87E-4	1.09E-7	1.96E-3	1.63E-7	2.93E-3
3b	6.30E+4	3.39E-9	2.13E-4	1.16E-8	7.32E-4	1.80E-8	1.14E-3
7	4.42E+5	1.58E-6	6.99E-1	1.58E-6	6.99E-1	1.58E-6	6.99E-1
8	5.88E+5	1.75E-8	1.03E-2	1.75E-8	1.03E-2	1.75E-8	1.03E-2
Total		2.10E-6	7.13E-1	2.10E-6	7.15E-1	2.10E-6	7.16E-1
Dose Rate from 3a and 3b (person-Rem/yr)			8.00E-4 (+8.1E-6)		2.69E-3 (+4.7E-5)		4.07E-3 (+1.1E-4)
Increase in Total Dose Rate (person-Rem/yr)	From 3 yr		—		1.73E-3 = 0.24% (+3.0E-5)		3.01E-3 = 0.42% (+9.84E-5)
	From 10 yr		—		—		1.28E-3 = 0.18% (+6.16E-5)
LERF from 3b (/yr)			3.39E-9 (+1.29E-10)		1.16E-8 (+7.50E-10)		1.80E-8 (+1.75E-9)
Increase in LERF (/yr)	From 3 yr		—		8.23E-9 (+6.21E-10)		1.47E-8 (+1.62E-9)
	From 10 yr		—		—		6.43E-9 (+1.00E-9)
CCFP %			76.48% (+0.006%)		76.87% (+0.060%)		77.18% (+0.100%)
Increase in CCFP	From 3 yr		—		0.39% (+0.05%)		0.70% (+0.09%)
	From 10 yr		—		—		0.31% (+0.04%)

(1) The numbers in parenthesis represent the incremental change (compared to Table 1) due to inclusion of the impact from the corrosion analysis.

Based on the results shown in Table 3, it can be seen that including corrosion effects in the ILRT assessment does not alter the conclusions from the original analysis. The increase in LERF from the 3-in-10 year interval to the 1-in-15 year interval is 1.47E-8/year, compared with 1.30E-8/yr without corrosion effects. This is still well below the Regulatory Guide 1.174 [3] acceptance criterion threshold for very small changes in risk of 1.0E-7/yr. This confirms that the proposed interval extension is acceptable from a risk basis. Additionally, the dose increase is 3.01E-03 person-rem/yr, which is only 0.4% above the 3-in-10 year value of 0.713 person-rem/yr. The increase in the CCFP is determined to be insignificant (77.18% for the 1-in-15 year case versus 76.48% for the 3-in-10 year case).

Table 4

QUAD CITIES STEEL LINER CORROSION SENSITIVITY CASES

Age (Step 3)	Containment Breach (Step 4)	Visual Inspection & Non-Visual Flaws (Step 5)	Increase in Class 3b Frequency (LERF) for ILRT Extension From 3 in 10 to 1 in 15 years (/yr)	
			Total Increase	Increase Due to Corrosion
Base Case (Doubles every 5 yrs)	Base Case (10% Walls, 1% Floor)	Base Case (10%)	1.47E-8	1.62E-9
Doubles every 2 yrs	Base	Base	1.60E-8	2.99E-9
Doubles every 10 yrs	Base	Base	1.41E-8	1.10E-9
Base	Base	15%	1.52E-8	2.16E-9
Base	Base	5%	1.41E-8	1.08E-9
Base	100% Walls, 10% Floor	Base	2.92E-8	1.62E-8
Base	1% Walls, 0.1% Floor	Base	1.32E-8	1.62E-10
Lower Bound				
Doubles every 10 yrs	1% Walls, 0.1% Floor	5%	1.31E-8	6.63E-11
Upper Bound				
Doubles every 2 yrs	100% Walls, 10% Floor	15%	5.50E-8	4.19E-8

Section 4
SUMMARY AND CONCLUSIONS

This analysis provides a quantitative assessment of the impact on risk of the potential for undetected steel liner corrosion due to an extension of the ILRT interval. The increase in LERF due to extending the test interval from 3 in 10 years to 1 in 15 years is 1.47E-8/yr, of which 1.62E-9/yr is due to corrosion. This value is considerably less than the RG 1.174 very small change criterion of 1.0E-7/yr. This confirms that the proposed interval extension is acceptable from a risk basis. Additionally, a series of parametric sensitivity studies regarding the potential age-related corrosion effects on the steel liner indicate that even with very conservative assumptions, the conclusions from the original analysis would not change; that is, the ILRT interval extension is judged to have a minimal impact on public risk and is therefore acceptable.

Section 5
REFERENCES

[1] *Quad Cities Risk Assessment to Support ILRT (Type A) Interval Extension Request*, prepared for Exelon by ERIN Engineering and Research, Inc., C46702044-5163, December 2002.

[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] *An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis*, Regulatory Guide 1.174, Revision 1, November 2002.

[4] *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.