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Jeremy G. Browning
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1CAN091401

September 2, 2014

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

SUBJECT: Response to Request for Additional Information Associated with Technical Specification Change to Extend the Type A Test Frequency to 15 Years
Arkansas Nuclear One, Unit 1
Docket No. 50-313
License No. DPR-51

REFERENCES;

1. Entergy letter to NRC, "License Amendment Request Technical Specification Change to Extend the Type A Frequency to 15 Years," dated December 20, 2013 (1CAN121302) (ML13358A195)

2. NRC letter to Entergy, "Arkansas Nuclear One, Unit No. 1 – Request for Additional Information Regarding License Amendment Request to Extend Integrated Leak Rate Testing Interval (TAC No. MF3279) (1CNA081402) (ML14209A085).

Dear Sir or Madam:

Pursuant to 10 CFR 50.90, Entergy Operations, Inc. (Entergy) requested an amendment to Arkansas Nuclear One, Unit 1 (ANO-1) Technical Specifications (TS). Specifically the proposed change would allow for the extension to the ten-year frequency of the ANO-1 Type A or Integrated Leak Rate Test that is required by TS 5.5.16 to be extended to 15 years on a permanent basis (Reference 1).

During their review of the request, the NRC has determined that additional information is needed to complete the review. Reference 2 provides the NRC's request for that additional information. The purpose of this submittal is to provide the requested information. See the attached responses.

The responses do not contain any new regulatory commitments. In addition, the responses do not alter the No Significant Hazards Consideration discussion provided in Reference 1.

If you have any questions or require additional information, please contact Stephenie Pyle at 479-858-4704.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on September 2, 2014.

Sincerely,

ORIGINAL SIGNED BY JEREMY G. BROWNING

JGB/rwc

Attachment: Response to Request for Additional Information License Amendment Request Proposing to Extend the Containment Integrated Leak Rate Testing Frequency

cc: Mr. Marc L. Dapas
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Attachment to

1CAN091401

Response to Request for Additional Information
License Amendment Request Proposing to Extend the
Containment Integrated Leak Rate Testing Frequency

**Response to Request for Additional Information
License Amendment Request Proposing to Extend the
Containment Integrated Leak Rate Testing Frequency**

By letter dated December 20, 2013 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML13358A 195), supplemented by letter dated March 11, 2014 (ADAMS Accession No. ML14070A399), Entergy Operations, Inc. (Entergy, the licensee), submitted a license amendment request (LAR) proposing a change to the Arkansas Nuclear One, Unit 1 (ANO-1) Technical Specifications (TSs). The proposed change would allow for the 10-year frequency of the ANO-1 Type A or Integrated Leak Rate Test (ILRT) that is required by TS 5.5.16, "Reactor Building Leakage Rate Testing Program," to be extended to 15 years on a permanent basis. In order for the U.S. Nuclear Regulatory Commission (NRC) staff to complete its review of the LAR, a response to the following request for additional information is requested.

1. **According to Regulatory Issue Summary 2007-06, "Regulatory Guide 1.200 Implementation," dated March 22, 2007 (ADAMS Accession No. ML070650428), the NRC staff expects that licensees fully address all scope elements with Revision 2 of Regulatory Guide (RG) 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," dated March 2009 (ADAMS Accession No. ML090140014), by the end of its implementation period (i.e., 1 year after the issuance of Revision 2 of RG 1.200). Revision 2 of RG 1.200 endorses, with exceptions and clarifications, the combined American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) probabilistic risk assessment (PRA) standard (ASME/ANS RA-Sa-2009).**

   (a) **In Section 4.5.2 of Attachment 1 to the LAR, the licensee stated that an industry peer review of the updated PRA model has been performed. Please clarify the scope of the peer review and the version of the ASME/ANS standard and RG 1.200 used for the peer review.**

   The ANO-1 PRA peer review was a full-scope review of the technical elements of the internal events at-power PRA. The 2009 peer review of the ANO-1 PRA used the Nuclear Energy Institute (NEI) 05-04 process and the ASME/ANS PRA Standard ASME/ANS RA-Sa-2009 (along with the Nuclear Regulatory Commission (NRC) clarifications provided in Regulatory Guide (RG) 1.200, Rev. 2). This peer review was performed on August 3 through 7, 2009.

   (b) **In Section 4.5.2 of Attachment 1 to the LAR, the licensee stated the ANO-1 internal events model has been updated to meet standards of RG 1.200, Revision 1, dated January 2007 (ADAMS Accession No. ML070240001). Given that the implementation date of RG 1.200, Revision 2, was April 2010 and the LAR was submitted in December 2013, if the peer review was not completed against RG 1.200, Revision 2, please describe any gaps between the peer review of the PRA**

model used in this application and RG 1.200, Revision 2, that are relevant to this submittal and also describe the impact of any gaps on this application.

The peer review was completed against RG 1.200, Revision 2 as shown in part (a) of this response. The inclusion of having ANO-1 internal events model to meet the RG 1.200, Revision 1, is an editorial error.

2. **Revision 2 of RG 1.200 endorses, with exceptions and clarifications, ASME/ANS RA-Sa-2009. In Regulatory Position 4.2 of RG 1.200, Revision 2, the NRC staff stated that it expects licensees to submit a discussion of the resolution of the peer review findings that are applicable to the parts of the PRA required for the application. The licensee stated in the LAR that an industry peer review of the updated PRA model has been performed.**

   (a) **For the PRA model used to support this application, please provide a list of Findings and Observations (F&Os) from the peer review relevant to this submittal.**

   See the enclosed matrix. This matrix is at the end of the responses.

   (b) **Please explain how these F&Os were addressed for this application and the impact of remaining open items on this application.**

   See the enclosed matrix. This matrix is at the end of the responses.

3. **The application refers to Electric Power Research Institute (EPRI) TR-1 009325, Revision 2-A, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals," dated October 2008 (ADAMS Accession No. ML 14024A045). EPRI TR-1009325, Revision 2-A, states, in part, that.**

   The most relevant plant-specific information should be used to develop population dose information. The order of preference shall be plant-specific best estimate, Severe Accident Mitigation Alternative (SAMA) for license renewal, and scaling of a reference plant population dose.

   (a) **Given that plant-specific population dose estimates were available as part of the ANO-1 SAMA analysis, please discuss the reasons for the decision to estimate the population dose based on scaling of Surry population doses.**

   To obtain the best basis for the ILRT dose estimates the election was made to use a more recent document than utilized in the SAMA assessment. The Emergency Plan (E-Plan), effective February 22, 2013, was selected over the SAMA population data. The SAMA population data was based on an estimate derived using information available in September 1999, Reference 15 from the SAMA documentation. Since the E-Plan is more recent, it was considered a better estimate for determining area

population. The E-Plan provided population estimates for 2016. The SAMA utilized 2025 estimates. A comparison is provided in the table below.

Table 1. Comparison of SAMA and ILRT Population Data

Source	Distance from Plant (miles)					Total
	10	20	30	40	50	
E-Plan Data	45,515	49,816	34,989	29,699	90,841	250,860
SAMA Data	58,648	41,458	51,272	48,170	127,871	327,419

Overall the population difference is 23%. The ILRT analysis assumes that 5% of the population is not evacuated or sheltered. The SAMA base case assumed all could be evacuated within 2.5 hours (135 minutes). The SAMA assessment did examine the 5% non-evacuation case. For that case it was reported that there was some increase for releases involving small leakage paths but overall the change was not significant.

A review of the SAMA basis documentation[1] identifies that the SAMA analysis method uses the same source for dose estimates as the ILRT. The mapping of the release classes is similar. Since the release classes were not recalculated, the election was made to utilize the most recent ANO-1 PRA information and the original data source for the evaluation. The selection of the most recent information was done in keeping with a desire to provide the most realistic estimation of the risk impact and to model the as-operated facility conditions.

(b) Please discuss whether using information from SAMA analysis would significantly change the estimated increase in population dose resulted from extending the Type A frequency.

The use of the SAMA population dose assuming the same conditions, evacuation at 135 minutes would potentially reduce the overall risk as represented by population dose. A comparison of the impact of using the SAMA population and assuming the 5% non-evacuation is provided in the table below in terms of percent increase in population dose associated with extending the ILRT test interval from the present 10 years to 15 years for the baseline analysis.

[1] **ANO-1 Level 3 Model Preparation, Revision 0, Entergy, 99-R-1007-02, November 1999.**

Table 2. Comparison of SAMA and ILRT Population Data on Population Dose

Population Source	Population Dose (person-rem/yr)			
	Baseline (10 year ILRT)	Increase Due to ILRT Extension	Total After Extension	% Change
ILRT Data	0.08990	0.00017	0.09007	N/A
SAMA Data	0.11651	0.00023	0.11674	29.61

The major change is a direct result of the difference in population estimates which reflects a different time period and would be expected to converge if the ILRT database were extrapolated to 2026. Even with this difference, the total increase is less than 30% which is considered to be a small change.

Since the source term data was taken from the same original source, the radionuclide composition would be unaltered by the selection of the SAMA data over the use of the direct source. Any differences would not be as a result of the source term data but rather changes in the base PRA model reflecting plant changes and updates to component performance. This would be reflected through the accident sequence class frequency.

4 EPRI TR-1009325, Revision 2-A states, in part, that where possible, the analysis should include a quantitative assessment of the contribution of external events (for example, fire and seismic) in the risk impact assessment for extended ILRT intervals. For example, where a licensee possesses a quantitative fire analysis and that analysis is of sufficient quality and detail to assess the impact, the methods used to obtain the impact from internal events should be applied for the external event ... This assessment can be taken from existing, previously submitted and approved analyses or another alternate method of assessing an order of magnitude estimate for contribution of the external event to the impact of the changed interval.

(a) Given that a Fire PRA (FPRA) model of ANO-1 has been used in the LAR to adopt National Fire Protection Association (NFPA) 805 performance based standard for fire protection submitted to NRC on January 29, 2014 (ADAMS Accession No. ML14029A438), please discuss the reasons that the FPRA model was not used to estimate the contribution of fire to large early release frequency (LERF) for this application.

At the time the ILRT analysis was developed the FPRA was not available, therefore, a surrogate was developed to estimate the potential contribution due to fire-induced initiating events (fire PRA). Since the completion of the ILRT study, the ANO-1 FPRA was submitted to the NRC in support of ongoing NFPA 805 implementation. A summary discussion of the potential impacts due to this change on the ILRT submittal is contained in the response to RAI 4(b) below.

(b) Please discuss whether using the FPRA model would significantly change the total estimated LERF.

The ILRT submittal used a surrogate for fire core damage frequency (CDF) that was developed by taking the internal events CDF (2.90E-6/yr) and subtracting out the LERF frequency associated with Interfacing System Loss of Coolant Accident (ISLOCA) and Steam Generator Tube Rupture (SGTR) (5.82E-8/yr). The resultant estimated CDF for fire initiating events is 2.84E-6/yr.

The current CDF for the ANO-1 FPRA is 5.95E-5/yr, and a change in Class 3b LERF is expected. To illustrate the potential change the key elements of the external events sensitivity study are reproduced below using the results from the FPRA in place of the surrogate values.

Per the guidance contained in the EPRI report[2] the figure-of-merit for the risk impact assessment of extended ILRT intervals is given as:

delta LERF = The change in frequency of Accident Class 3b

An approximation for the early CDF applicable to EPRI Accident Class 3b is developed by taking the total CDF and subtracting out the LERF-related frequency. Since this information is only available for the fire contribution to external events, the seismic external events conservatively does not subtract any LERF contribution away from the surrogate value. This process is outlined below.

CDF_{FIRE} = 5.95E-5/yr – 5.15E-6/yr = 5.44E-5/yr

$CDF_{SEISMIC}$ = 2.4E-6/yr + 1.03E-7/yr = 2.5E-6/yr

Class 3b Frequency = [(CDF_{FIRE}) + ($CDF_{SEISMIC}$)] * Class 3b Leakage Probability

Class 3b Frequency = [(5.44E-5/yr) + (2.5E-6/yr)] * 2.3E-03 = 1.31E-7/yr

As previously stated no adjustment is made to the seismic CDF value since LERF sequences are typically associated with SGTR or ISLOCA sequences which are not represented by the external event assessments. This is potentially conservative, but is reasonable based on the simplified assessment, the conservative nature of the external events studies and the fact that many of the external event scenarios are long term station blackout and long-term level of analysis detail. The change in LERF is estimated for the one in ten year and one in 15-year cases, and the change is defined for the external events in Table 3.

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[2] **Gisclon, J. M., et al, Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals: Revision 2-A of 1009325, Electric Power Research Institute, 1018234, October 2008.**

Table 3. ANO-1 Upper Bound External Event Impact on ILRT LERF Calculation

Hazard	EPRI Accident Class 3b Frequency			LERF Increase (from 1 per 10 years)
	3 per 10-year	1 per 10-year	1 per 15-year	
External Events	1.31E-7	4.36E-7	6.54E-7	2.18E-7
Internal Events	6.56E-9	2.19E-8	3.28E-8	1.09E-8
Combined	1.37E-7	4.58E-7	6.87E-7	2.29E-7

The internal event results are also provided to allow a composite value to be defined. When both the internal and external event contributions are combined the total change in LERF no longer meets the criterion for a very small change (less than 1E-7/yr). According to RG 1.174 the combined delta LERF value of 2.29E-7/yr is considered small since it falls between 1E-7/yr and 1E-6/yr. When an increase in LERF is classified as small the supporting requirement is that the total LERF must be less than 1E-5/yr.

The total LERF from the ILRT submittal is the summation of the EPRI Classifications 2, 3b, and 8. The information above in Table 3 pertains to the Class 3b contribution. The total internal events LERF using information contained in Table 13 of the ILRT analysis is calculated below.

$LERF_{Internal}$ = Class 2 + Class 3b + Class 8

$LERF_{Internal}$ = 2.94E-9/yr + 3.28E-8/yr + 3.97E-8/yr = 7.55E-8/yr

Since the accident progression for external events LERF would be similar to that of the internal events, a ratio (R) is developed to generate the remaining external event LERF contributors (Class 2 and Class 8). By combining the internal events Class 2 and Class 8 frequencies and dividing by the total CDF the relationship between these LERF contributors and total CDF is established. This process is outlined below.

R = (Class 2 + Class 8) / $CDF_{Internal}$

R = (2.94E-9/yr + 3.97E-8/yr) / 2.90E-6/yr = 1.47%

The external events LERF is comprised of two different phenomena (fire and seismic). The CDF from fire is significantly higher than the internal events CDF, and the seismic CDF is almost equal to the internal events CDF. Therefore, the resultant LERF from both of these external events combined should be higher than the internal events. Multiplying the external events CDF by the result from above (1.47%) generates the remaining LERF (Class 2 and Class 8). Combining all the external event LERF classes yields a total approximation for the external events LERF.

$LERF_{External}$ = ($CDF_{External}$ * R) + Class 3b

$LERF_{External}$ = (5.69E-5/yr * 0.0147) + 6.54E-7 = 1.49E-6/yr

Therefore the total combined internal and external event LERF is 1.57E-6/yr. This value meets the criteria set forth in RG 1.174 of being less than 1.0E-5//yr. It should be noted that these results are dominated by the contribution from the external events LERF. This is expected and also confirms the previously calculated values for Class 3b.

The inclusion of the FPRA results did cause an increase in the contribution to risk from Class 3b associated the Type A testing extension. This increase was sufficient for the net change to transition above 1.0E-7/yr and is no longer classified as very small by the definitions contained in RG 1.174. With the FPRA information included the change is now reclassified as small. This is not a significant transition as long as the overall LERF contribution from all sources is less than 1.0E-5/yr. This criterion is met and the total LERF is estimated at 1.57E-6/yr. While the FPRA results did slightly alter the external events sensitivity study, the criteria of RG 1.174 remains satisfied and does not invalidate the baseline analysis.

5. **Section 5.1.5.1 of EPRI TR-1009325, Revision 2-A uses the Calvert Cliffs methodology in evaluating the impact of liner corrosion on the extension of ILRT testing intervals. This assessment was based on two observed corrosion events at North Anna Power Station, Unit 2, and Brunswick Steam Electric Plant, Unit 2. As there have been additional instances of liner corrosion that could be relevant to this assessment, please provide a more complete accounting of all observed corrosion events relevant to ANO-1 containment, and an evaluation of the impact on risk results when all relevant corrosion events are included in the risk assessment.**

Since Calvert Cliffs performed their evaluation of liner corrosion likelihood in 2002, more liner corrosion events have been observed. An evaluation of the impact on risk using the adopted methodology but extending the data collection period until December 31, 2013, is conducted below. Therefore, the new data collection period begins in September of 1996 and ends on December 31, 2013. The extended collection period increase the total data collection time from 5.5 years to 17.25 years.

Over the 17.25 years, more containment liner corrosion events occurred. In 2011, the NRC published a technical letter report that analyzed containment liner corrosion events occurring at operating nuclear power plants in the United States[3]. The results of this analysis were five containment liner corrosion events in almost 15 years at 66 possible sites in the United States. Two of the five events are the same existences of corrosion used by Calvert Cliffs in their liner corrosion analysis (North Anna Power Station Unit 2 and Brunswick Steam Electric Plant Unit 2). The next event took place at D.C. Cook Unit 2 in March of 2001. A small hole was discovered in the liner plate that the plant suspected was man-made. In 2009, a through-wall penetration caused by a piece of wood embedded in the concrete was identified at Beaver Valley. It should be noted that in 2006 during the Beaver Valley Unit 1 steam generator replacement surface corrosion was identified. This corrosion had yet to cause penetration in the liner, but since the discovery of this corrosion occurred during a

[3] **Dunn, D. S., et al, <u>Containment Liner Corrosion Operating Experience Summary Technical Letter Report – Revision 1</u>, USNRC, August 2011.**

steam generator replacement and not a normal inspection, the event was included with the conservative assumption that the corrosion would have been discovered after it penetrated the steel liner. The last event occurred in the fall of 2013 at Beaver Valley Unit 1[4]. Thus over the 17.25-year data collection period six liner corrosion events occurred at a possible 66 plant locations.

Table 4 summarizes the results obtained from the Calvert Cliffs methodology utilizing a more recent data collection period.

Table 4. ANO-1 Liner Corrosion Risk Assessment Results Using CCNP Methodology with an Extended Data Collection Period

Step	Description	Containment Cylinder and Dome (85%)		Containment Basemat (15%)	
1	Historical liner flaw likelihood Failure data: containment location specific Success data: based on 70 steel-lined containments and 5.5 years since the 10CFR 50.55a requirements of periodic visual inspections of containment surfaces	Events 6 6 / (66 x 17.25) = 5.27E-3/yr		Events: 0 Assume a half failure 0.5 / (66 x 17.25) = 4.39E-4/yr	
2	Aged adjusted liner flaw likelihood During the 15-year interval, assume failure rate doubles every five years (14.9% increase per year). The average for the 5th to 10th year set to the historical failure rate.	Year 1 average 5-10 15	Failure rate 2.14E-3/yr 5.27E-3/yr 1.49E-2/yr	Year 1 average 5-10 15	Failure rate 1.78E-4/yr 4.39E-4/yr 1.24E-3/yr
		15 year average = 6.42E-3/yr		15 year average = 5.58E-4/yr	
3	Increase in flaw likelihood between 3 and 15 years Uses aged adjusted liner flaw likelihood (Step 2), assuming failure rate doubles every five years.	0.74% (1 to 3 years) 4.24% (1 to 10 years) 9.63% (1 to 15 years)		0.06% (1 to 3 years) 0.36% (1 to 10 years) 0.84% (1 to 15 years)	

[4] **Sepelak, B., Containment Liner Through Wall Defect Discovered During Planned Visual Inspection, FirstEnergy Nuclear Operating Company (FENOC), LER 2013-002-01, February 2014.**

Step	Description	Containment Cylinder and Dome (85%)	Containment Basemat (15%)
4	Likelihood of breach in containment given liner flaw	1%	0.1%
5	Visual inspection detection failure likelihood	10% 5% failure to identify visual flaws plus 5% likelihood that the flaw is not visible (not through-cylinder 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 x 4 x 5)	0.00074% (3 years) 0.74% x 1% x 10% 0.00424% (10 years) 4.24% x 1% x 10% 0.00963% (15 years) 9.63% x 1% x 10%	0.00006% (3 years) 0.06% x 0.1% x 100% 0.00036% (10 years) 0.36% x 0.1% x 100% 0.00084% (15 years) 0.84% x 0.1% x 100%

The total likelihood of the corrosion-induced, non-detected containment leakage is the sum of Step 6 for containment cylinder and dome and the containment basemat.

Total likelihood of non-detected containment leakage (3 yr) = 0.00074% + 0.00006% = 0.0008%

Total likelihood of non-detected containment leakage (10 yr) = 0.00424% + 0.00036% = 0.0046%

Total likelihood of non-detected containment leakage (15 yr) = 0.00963% + 0.00084% = 0.01047%

This likelihood is then multiplied by the non-LERF containment failures. This value is calculated by the following equation for each period of interest. LERF is comprised of Class 2, Class 8, and Class 3b cases as shown below.

Non-LERF = CDF – Class 2 – Class 8 – Class 3b

Table 5 presents the data and the resultant increase in LERF due to liner corrosion for each case.

Table 5. Liner Corrosion LERF Adjustment Using CCNP Methodology

Case	CDF (/yr)	Class 2 (/yr)	Class 8 (/yr)	Class 3b (/yr)	Likelihood of Non-detected Corrosion Leakage	Increase in LERF (/yr)
3-years	2.90E-6	2.94E-9	3.97E-8	6.56E-9	8.00E-6	2.28E-11
10-years	2.90E-6	2.94E-9	3.97E-8	2.19E-8	4.60E-5	1.31E-10
15-years	2.90E-6	2.94E-9	3.97E-8	3.28E-8	1.05E-4	2.96E-10

This contribution is added to the Class 3b LERF cases and the sensitivity analysis performed. Table 6 provides a summary of the base case as well as the corrosion sensitivity case. The “Delta Person-Rem” column provides the change in person-rem between the case without corrosion and the case that considers corrosion. Values within parentheses “( )” indicate the change or delta between the without corrosion and corrosion cases.

Table 6. ANO-1 Summary of Base Case and Corrosion Sensitivity Cases

EPRI Class	Base Case (3 per 10 years)					1 per 10 years					1 per 15 years				
	Without Corrosion		With Corrosion			Without Corrosion		With Corrosion			Without Corrosion		With Corrosion		
	Frequency	Person-rem per year	Frequency	Person-rem per year	Delta Person-Rem per year	Frequency	Person-rem per year	Frequency	Person-rem per year	Delta Person-Rem per year	Frequency	Person-rem per year	Frequency	Person-rem per year	Delta Person-Rem per year
1	2.62E-6	3.01E-4	2.62E-6	3.01E-4	-2.63E-9	2.54E-6	2.92E-4	2.54E-6	2.92E-4	-1.50E-8	2.49E-6	2.86E-4	2.48E-6	2.86E-4	-3.41E-8
2	2.94E-9	1.21E-3	2.94E-9	1.21E-3	n/a	2.94E-9	1.21E-3	2.94E-9	1.21E-3	n/a	2.94E-9	1.21E-3	2.94E-9	1.21E-3	n/a
3a	2.64E-8	3.03E-5	2.64E-8	3.03E-5	n/a	8.79E-8	1.01E-4	8.79E-8	1.01E-4	n/a	1.32E-7	1.52E-4	1.32E-7	1.52E-4	n/a
3b	6.56E-9	7.55E-5	6.59E-9	7.58E-5	2.63E-7	2.19E-8	2.52E-4	2.20E-8	2.53E-4	1.50E-6	3.28E-8	3.77E-4	3.31E-8	3.81E-4	3.41E-6
6	4.97E-11	2.09E-6	4.97E-11	2.09E-6	n/a	4.97E-11	2.09E-6	4.97E-11	2.09E-6	n/a	4.97E-11	2.09E-6	4.97E-11	2.09E-6	n/a
7	2.12E-7	5.34E-2	2.12E-7	5.34E-2	n/a	2.12E-7	5.34E-2	2.12E-7	5.34E-2	n/a	2.12E-7	5.34E-2	2.12E-7	5.34E-2	n/a
8	3.97E-8	3.46E-2	3.97E-8	3.46E-2	n/a	3.97E-8	3.46E-2	3.97E-8	3.46E-2	n/a	3.97E-8	3.46E-2	3.97E-8	3.46E-2	n/a
CDF	2.90E-6	8.97E-2	2.90E-6	8.97E-2	2.60E-7	2.90E-6	8.99E-2	2.90E-6	8.99E-2	1.49E-6	2.90E-6	9.01E-2	2.90E-6	9.01E-2	3.37E-6
Class 3b LERF	6.56E-9		6.59E-9 (2.28E-11)			2.19E-8		2.20E-8 (1.31E-10)			3.28E-8		3.31E-8 (2.96E-10)		
Delta LERF (from base case of 3 per 10 years)						1.53E-8		1.53E-8 (1.08E-10)			2.63E-8		2.65E-8 (2.73E-10)		
Delta LERF from 1 per 10 years						N/A					1.09E-8		1.11E-8 (1.66E-10)		

The expanded scope data collection period for liner corrosion does not result in an increase in LERF sufficient to invalidate the baseline analysis and the overall impact of corrosion inclusion is negligible.

6. **The LAR states that there is one primary containment surface associated with the area around the equipment hatch that requires augmented examinations in accordance with the ASME Boiler and Pressure Vessel Code (ASME Code) Section XI, IWE-1240. Please provide information regarding the findings that led to the augmented examination. Also, please provide information that would demonstrate proper and effective monitoring and managing of this condition.**

During an ASME Section XI IWE-required general visual examination of the ANO-1 Equipment Hatch (C-1) during refueling outage 1R18 in 2004, an area of pitting and corrosion on the equipment hatch door and flange area was documented. This area was cleaned and repainted.

A review of the examination findings concluded that although the area was acceptable by examination in accordance with IWE-3122.1, the area was subject to accelerated degradation and the requirements of IWE-1240 were to be implemented. The Containment Inservice Inspection (CISI) program was revised to include augmented examinations in accordance with IWE-1240 and Table IWE-2500. In addition, it was determined that repeated wetting and submergence due to trapping storm water against the exterior of the equipment hatch was responsible for the pitting and corrosion. At the conclusion of each refuel or forced outage actions are taken to adequately seal the Kelly closure above the equipment hatch prevent/minimize the intrusion of rain water that can accumulate against the equipment hatch.

Subsequent to the IWE examination conducted in 2004, augmented examinations of the equipment hatch have been performed in accordance with ASME Table IWE-2500 during refueling outages 1R19 (2005), 1R22 (2010) and 1R24 (2013). Results of the 1R19 examination identified pitting and corrosion on the surfaces of the equipment hatch door and flange area. Ultrasonic thickness measurements were taken. An average of 1.0-inch thickness was recorded. As with the 2004 examination, the acceptance criteria of IWE-3122.1 were satisfied and the operability determination concluded that the structural integrity of the ANO-1 Reactor Building and Equipment Hatch were not impacted and that the system remained operable. Corrective actions to clean and re-coat the affected surfaces were completed in 2006.

The satisfactory completion of the augmented examinations conducted during 1R22 and 1R24 continued to provide assurance of structural integrity and compliance with the acceptance standards.

The final augmented examinations of the ANO-1 Equipment Hatch to be conducted in this fourth CISI interval are scheduled to be performed during the refueling outage of 1R26 (2016). Upon satisfactory completion of that examination and if it can be determined that the examination area is no longer susceptible to accelerated degradation per IWE-1240, the CISI program will be revised to discontinue the augmented examinations of IWE Table IWE 2500.

7. **Attachment 4 of the LAR, Tables 4-2 and 4-3, include a brief description of the results of reactor building interior and exterior structural inspections and ASME Code, Section XI, Subsection IWE inspections. Both tables indicate that numerous deficiencies were noted; however, they do not include details regarding these deficiencies. Please discuss highlights of the significant findings from the ASME Code, Section XI, Subsection IWE and IWL examinations performed since the last Type A test on the containment pressure-retaining structures and components, in accordance with the ANO-1 containment in-service inspection (CISI) program, and actions taken to disposition them. In the response, provide information that would demonstrate proper and effective implementation of the ANO-1 CISI program in monitoring and managing degradation to ensure that containment structural and leak-tight integrity has been, and will continue to be, maintained through the service life of the plant. The response should include relevant highlights of examinations performed on the containment penetrations (with seals, gaskets, and bolted connections), the containment steel liner, moisture barrier, and the reinforced concrete containment structure. Also, please discuss highlights of findings from recent inspections from the ANO-1 containment coating inspection program and actions taken to disposition them.**

   The most recent Type A test on the ANO1 containment pressure system was conducted in 2005 during refueling outage 1R19.

   Review of the IWE/IWL examinations including findings and corrective actions performed since the last Type A test concludes that although there were numerous deficiencies identified and summarized in Table 1 below, no flaws or areas of degradation exceeding the allowable acceptance standards of IWE-3500 or IWL-3200 were identified.

   The following table is a compilation of the examination results from 2005 to 2014.

Table 7
IWE/IWL Examination Results1R19 thru 1R24 Refuel Outages

Area Examined	Deficiency	Disposition
Containment Equipment Hatch	Pitting, Corrosion	Acceptable without engineering evaluation or repair / replacement.
Containment Steel Liner – Dome Sections	Minor rust, blistering, and flaking	Acceptable without engineering evaluation or repair / replacement.
Containment Steel Liner – Wall Sections	Minor paint blistering	Acceptable without engineering evaluation or repair / replacement.
Containment Steel Liner – Reactor	Light rust	Acceptable without engineering evaluation or repair / replacement.

Area Examined	Deficiency	Disposition
Building Sump Penetrations		
35 Year IWL Tendon Examination	The examination report states while no indications were found that challenge current structural integrity or leak tightness of the containment, three indications were found that require evaluation under IWL-3300: 1) Protruding button heads on Tendon 31H28. 2) Change in size of existing concrete cracks at bearing plate of Tendon 31H08. 3) Tendons 31H21 and 31H28 accepted more than 10% of total grease capacity.	1) Tendon 31H28 was evaluated to still be operable. Calculations show that a total of seven wires can be missing from each tendon and remain acceptable. 2) This condition has been determined to be acceptable as is, and tendon 31H08 continues to perform its design function. The two cracks were originally identified in 1999. Difference in reported crack width is due to heat, erosion of the concrete at the crack, and inspectors. 3) The acceptance of greater than 10% of the net duct volume of corrosion inhibitor for tendons 31H21 and 31H28 is accepted by evaluation. No additional examinations are required for this condition: however, these tendons will be further examined during the next scheduled IWL tendon surveillance.

Area Examined	Deficiency	Disposition
40 Year IWL Tendon Examination	The examination report states while no indications were found that challenge current structural integrity or leak tightness of the containment, four indications were found that require evaluation under IWL-3300. Additionally, the observed indications do not show the presence of degradation in inaccessible areas. 1) Tendon 3D104 did not meet elongation requirements during retensioning. 2) Tendon 31H05 was found with one missing buttonhead. 3) Three protruding wires on Tendon 31H19. 4) Two cracks wider than 0.010" were detected at tendon 31H08.	1) An evaluation of the variation was completed considering the testing of a tendon wire, the tendon liftoff values, and other factors, and the condition was found to be acceptable without further evaluation or repair/replacement activities. 2) The original design bases calculation determined a total of seven wires can be missing, removed, or broken from a tendon and still be acceptable. 3) Per the original design bases calculation, a total of seven wires can be missing from each tendon and still be considered acceptable. Using this conservative approach, the tendon still meets its design requirements, and remains effective in maintaining the required post-tensioned force. 4) Due to lack of evidence of corrosion of the steel reinforcement, margin provided by the robust concrete reinforcing, and length of time of the potential exposure of the steel, it was determined that the steel reinforcing is currently not significantly affected and the buttress itself maintains its design capacity. Both of these contributors to cracking can be eliminated by sealing the area from moisture and oxygen. Using Belzona, a non-structural grouting and sealing repair (reference ACI 224.1R-07 Section 3.3) of the crack was implemented.

1R21 (2008) Coatings Assessment Inspection - 1R21 was the first outage to implement the coatings assessment program. New areas of degraded qualified coatings and unqualified coatings were identified during the coatings assessment walkdowns. Each area of newly identified degraded qualified coatings is categorized in accordance with procedures based on the extent of degradation. The categories as defined by procedures are as follows;

A. Immediate Repair Required – These are sizeable areas of degraded coatings with the potential to exceed margin limits by a serious degradation of equipment in the next cycle. This may include removal of the degraded or defective coating by scraping chips, flakes, blisters, etc. back to sound coating and allow repair at the next outage.

B. Repair At Next Refueling Outage – These are areas typically of minor to moderate mechanical damage that do not pose a threat to further degrade or generate a debris source in the event of a Design Basis Accident but do need to be repaired.

C. Monitor And Trend – These are areas with very minor defects that do not pose a threat for generation of debris and do not warrant repair.

D. Satisfactory – These are areas of good coatings in satisfactory condition that do not possess any notable defects.

No coatings requiring "Immediate Repair" were identified. The newly identified areas of degraded qualified coatings during 1R21 exhibited flaking and/or blistering. These areas were small, isolated areas and did not portray an overall general failure of the coating of the Reactor Building. In order to preserve as much coatings margin as possible, these identified coatings are classified as "Repair at Next Refueling Outage." In addition, these areas are added to the overall Degraded Qualified Coatings Log.

1R22 (2010) Coatings Assessment Inspection - Coating walkdowns were conducted during 1R22. Multiple areas of degraded qualified and unqualified coatings were identified during the walkdowns.

Due to the change in acceptable ZOI (Zone-of-Influence) for primer-only coatings per NRC guidance, the north and south steam generator cavities were also walked down to identify and document areas with a primer-only coating. During 1R22, new areas of unqualified coatings were identified and added to the Unqualified Coatings Log. New areas of degraded qualified coatings were also identified; however, these coatings were removed during 1R22 and therefore, were not added to the Degraded Coatings Log. Some previously identified degraded and unqualified coatings were repaired and replaced during 1R22 and therefore, these areas have been removed from the coatings logs.

1R23 (2011) Coatings Assessment Inspection - Coating walkdowns of the accessible areas at each elevation of the Reactor Building was conducted during 1R23. Multiple new areas of degraded qualified coatings were identified, but there were no gross coating failures identified, and only minor isolated deficiencies were noted. New areas of unqualified coatings in the form of graffiti/spray paint were identified at various locations of the Reactor Building. Ten items on the unqualified coatings log were repaired under various work orders subsequently; therefore, these entries were removed from the logs.

1R24 (2013) Coatings Assessment Inspection - No coatings assessment inspections were performed during 1R24. Multiple new areas of degraded qualified coatings were identified but there were no gross coating failures identified, and only minor isolated deficiencies were noted. No repairs to degraded/unqualified coatings were made.

8. **Please provide the schedule of inspections, including the corresponding refueling outage, that were, or will be, performed on the containment structure in accordance**

with ASME Section XI, Subsection IWE and IWL, and explain how it meets the provisions in Section 9.2.3.2 of Nuclear Energy Institute (NEI) 94-01, Revision 2-A, "Industry Guideline for Implementing the Performance-Based Option of 10 CFR Part 50, Appendix J," and Condition 2 in Section 4.1 of the NRC safety evaluation dated June 25, 2008 (ADAMS Accession No. ML0811401 05) for topical report NEI 94-01, Revision 2-A.

Section 9.2.3.2 of NEI 94-01, Revision 2-A, and Condition 2 in Section 4.1 of the NRC safety evaluation for topical report NEI 94-01, Revision 2-A require supplemental general visual inspections of accessible interior and exterior surfaces of the containment for structural deterioration that may affect the containment leak-tight integrity. These inspections must be conducted prior to each Type A test and during at least three other outages before the next Type A test if the interval for the Type A test has been extended to 15 years.

The ANO-1 CISI program contains the CISI Interval schedule of examinations. Table 8 summarizes that schedule.

Table 8
Containment Examination Schedule

Examination Type	Refuel Outage	Dates
30-Yr. IWL Containment Inspection	1R17/1R18	2002-2004
ILRT Type A Test	1R19	2005
IWE Containment Liner Dome Sections and Moisture Barrier	1R19	2005
IWE Containment Liner Dome and Barrel Sections	1R20	2007
IWE Containment Liner Dome Sections and Moisture Barrier	1R21	2008
IWE Containment Liner Barrel Sections	1R22	2010
35-Yr. IWL Containment Inspection	1R20/1R21	2007-2009
IWE Containment Liner Dome Sections	1R23	2011
IWE Containment Liner Barrel Sections	1R24	2013
40-Yr. IWL Containment Inspection	1R24/1R25	2012-2014
IWE Containment Liner Dome Sections and Moisture Barrier	1R25	2014
IWE Containment Liner Barrel Sections	1R26	2016
45-Yr. IWL Containment Inspection	1R27/1R28	2017-2019
ILRT Type A Test	1R29	2020
50-Yr. IWL Containment Inspection	1R30/1R31	2022-2024
55-Yr. IWL Containment Inspection	1R34/1R35	2027-2029
60-Yr. IWL Containment Inspection	1R37/1R38	2032-2034

9. **Please provide information of instances during implementation of the ANO-1 CISI program in accordance with ASME Code, Section XI, Subsections IWE/IWL, where existence of or potential for degraded conditions in inaccessible areas of the concrete containment structure and steel liner were identified and evaluated based on conditions found in accessible areas, as required by Title 10 of the Code of Federal Regulations (10 CFR), Part 50, paragraph 55a(b)(2)(viii)(E) and 10 CFR 50.55a(b)(2)(ix)(A). If there were any instances of such conditions, discuss the findings and corrective actions taken to disposition the findings.**

   No areas of potential degraded conditions in inaccessible areas of the concrete containment structure and steel liner have been identified based on conditions found in accessible areas. See the summaries of the IWE/IWL examinations below. Information on the equipment hatch augmented examinations is supplied in an earlier response and is not repeated here.

   1R20 (2007) IWE EXAMINATION – It was identified that the top coat of the liner plate was failing, causing elongated blistering. The coating blisters were flaking off the liner plate resulting in loose deleterious material. The exposed primer coat (zinc) did not show any appreciable deteriorating. This condition was previously identified in 2001.

   The evaluation concluded that the deficient coating found did not impact the structural integrity of the reactor building liner plate. The IWE inspection for 1R16 found the primer intact providing full protection to the liner plate.

   1R21 (2008) IWE EXAMINATION –The condition noted in the examination was essentially the same as the previous inspection with no new areas identified from 1R19 to 1R21. The identified areas have light rust with no visible pitting. Rusting of this type and level did not significantly affect the thickness of the liner plate to perform its function and is acceptable.

   The acceptance review indicated that the conditions noted on the surface area did not jeopardize the structural integrity of the containment. The conditions were acceptable without engineering evaluation or repair/replacement.

   1R22 (2010) IWE EXAMINATION – The top coat of the liner plate is failing, causing elongated blistering. The acceptance review states the conditions noted in this examination are essentially the same as the previous inspections. The identified areas had elongated blistering exceeding a size 5 with the top coat flaking off but the zinc primer still intact, and light surface rust greater than a grade 7 along the liner seam welds with no visible pitting.

   The zinc primer was still intact protecting the liner plate. This indicates that the liner plate thickness was not being affected allowing the liner plate to perform its function and was acceptable without engineering evaluation or repair/replacement.

   Rusting of this type (Type 1) and level does not significantly affect the thickness of the liner plate to perform its function and was acceptable without engineering evaluation or repair/replacement.

1R23 (2011) IWE EXAMINATION - The condition noted in the examination was essentially the same as the previous inspection with no new areas identified. The identified areas had light rust with no visible pitting. Rusting of this type and level did not significantly affect the thickness of the liner plate to perform its function and is acceptable.

The acceptance review indicated that the conditions noted on the surface area did not jeopardize the structural integrity of the containment. The conditions were acceptable without engineering evaluation or repair/replacement.

The Moisture Barrier examination was acceptable with no defect noted.

1R24 (2013) IWE EXAMINATION – During this examination the top coat of the liner plate was identified as failing, causing elongated blistering. A review of the examination reports accepting these conditions for this outage noted they were essentially the same as the previous inspections. The identified areas had elongated blistering exceeds a size 5 with the top coat flaking off but the zinc primer still intact, and light surface rust greater than a grade 7 along the liner seam welds with no visible pitting.

35th Year ANO-1 CONTAINMENT BUILDING TENDON SURVEILLANCE AND CONCRETE INSPECTION ENGINEERING EVALUATION REPORT (2007 – 2009) states in the summary of results that observed indications did not show the presence of degradation in inaccessible areas.

40th Year ANO-1 CONTAINMENT BUILDING TENDON SURVEILLANCE AND CONCRETE INSPECTION ENGINEERING EVALUATION REPORT (2012 – 2014) states in the summary of results that observed indications did not show the presence of degradation in inaccessible areas.

10. **As stated in ANO-1 Safety Analysis Report, Section 5.2.2.1.4, "Special Penetrations," expansion joint bellows at the fuel transfer tube provide for the relative movement between the reactor building, internals, and the auxiliary building. For any bellows used on penetrations through containment pressure retaining boundaries at ANO-1, please provide information on their location, inspection, testing and operating experience with regard to detection of bellows leakage.**

    The ANO-1 fuel transfer tube is connected to the Reactor Building liner via a fixed flued welded pipe connection. A bellows assembly is provided on the Auxiliary Building fuel tilt pit side to allow for differential expansion between the Auxiliary Building and the Reactor Building. Another bellows assembly is provided on the Reactor Building side to allow for differential expansion on the Reactor Building side of the flued-head connection and to provide a seal between the refueling canal and the containment at the transfer tube penetration.

ANO-1 credits the flued head, the transfer tube and the transfer tube blind flange (closure plate) on the Reactor Building side for the containment pressure boundary during plant operation and does not credit the Reactor Building bellow for this function. The closure plate has O-ring seals which can be tested.

One purpose of the Reactor Building side bellows is to provide a hydraulic boundary for the fuel transfer canal, and any leakage would be evident in the containment outside the canal. The bellows on the Auxiliary Building side provides a hydraulic boundary between the fuel tilt pit and the Auxiliary Building/Reactor Building gap area.

This discussion is consistent with ANO-1 SAR Sections 5.2.2.1.4, 9.4.2.4 and 9.4.2.5. ANO-1 SAR Section 9.6.1.6 notes that the "portion of the transfer tube inside the reactor building is an integral part of the reactor containment during reactor operation". ANO-1 SAR Section 9.6.2.3 states:

> During reactor operation, a bolted and gasketed closure plate, located on the reactor building end of the fuel transfer tube, prevents leakage of water from the spent fuel storage pool into the transfer canal in the event of a leak through the fuel transfer tube valve. Both the spent fuel storage pool and the fuel transfer canal are completely lined with stainless steel for leak tightness and ease of decontamination. The fuel transfer tube is appropriately attached to these liners to maintain leak integrity.

11. Attachment 5 to the LAR contains a summary table of components that did not meet the administrative limit for Type B and Type C testing. Please describe the causes and corrective actions taken to address the components that did not demonstrate acceptable performance in accordance with the ANO-1 Reactor Building Leakage Rate Testing Program.

Below is a discussion of apparent causes for components listed in Attachment 5 of the LAR. It should be noted that ANSI 56.8 only requires apparent cause determination when a component is above its Appendix J administrative limit. As noted in the original LAR, several local leak rate tests (LLRTs) were listed because a component had failed an operational pre-limit not its Appendix J administrative limit. The discussion below only addresses components that have failed their Appendix J administrative limit.

SV-1818/PSV-1800 discussion:

<u>October 8 and 19, 2002 tests</u> - SV-1818/PSV-1800 failed its as-found October 8, 2002, and its as-left LLRT on October, 19, 2002. The administrative limit for this test is 2000 standard cubic centimeter per minute (sccm). It was determined that PSV-1800 was leaking. PSV-1800 was replaced. It was noted that PSV-1800 exhaust port was inspected, and no leakage was determined. Also, it was verified that no leakage existed through the boundary valves. It was recommended that the valve be worked next refueling outage.

<u>May 6, 2004 test</u> - SV-1818/PSV-1800 passed their as-found LLRT with a leakage rate of 230 sccm against an administrative limit of 2000 sccm. The valves passed their as-found

LLRT. The valves subsequently failed their as-left LLRT. The valves were scheduled to be worked the next outage.

October 14, 2005 test - SV-1818/PSV-1800 failed their as-found LLRT with a leakage rate of 3400 sccm against an administrative limit of 2000 sccm. SV-1818 was cut out and replaced. An as-left LLRT was performed on December 3, 2005, with satisfactory results (19 sccm against an administrative limit of 2000 sccm). No description was recorded about the valve condition.

April 28, 2007 test - SV-1818/PSV-1800 failed their as-found LLRT with a leakage rate of 5800 sccm against an administrative limit of 2000 sccm. No record of maintenance was found and the valve was as-left tested on the same day, so it is assumed that the valve was flushed of debris and retested. The apparent cause was debris on the valve seat.

March 24, 2010 test - SV-1818/PSV-1800 failed their as-found LLRT with a leakage rate of 4850 sccm against an administrative limit of 2000 sccm. SV-1818 was disassembled and found that the disc was pitted, disc guide scored / scratched and debris was covering the valve internals. The disc and disc guide were replaced and the valve body seat lapped. An as-left LLRT was performed on April 14, 2010, with satisfactory results (5 sccm against an administrative limit of 2000 sccm). The apparent cause was debris on the valve seat which damaged the valve disc.

October 20, 2011 and November 15, 2011 tests - SV-1818/PSV-1800 failed their as-found LLRT with a leakage rate of 10500 sccm against an administrative limit of 2000 sccm. The valve disc was replaced and the valve body seat was lapped on November 8, 2011. PSV-1800 was replaced on October 22, 2011. SV-1818/PSV-1800 failed their as-left LLRT with a leakage rate of 3500 sccm against an administrative limit of 2000 sccm. The apparent cause for SV-1818/PSV-1800 leakage was disc damage.

May 3, 2013 test - SV-1818/PSV-1800 failed their as-found LLRT with a leakage rate of 5900 sccm against an administrative limit of 2000 sccm. The valve was disassembled and replacement parts were replaced on June 6, 2013. An as-left LLRT was performed on June 11, 2013, with satisfactory results (168 sccm against an administrative limit of 2000 sccm).

SV-1840 discussion: SV-1840 failed its as-found LLRT on October 8, 2002. Maintenance was performed October 12, 2002. Disc damage was noted, and the disc was replaced. An as-left LLRT was performed on October 19, 2002, with satisfactory results (430 sccm against an administrative limit of 2000 sccm). The apparent cause for SV-1840 leakage was disc damage.

CV-4804 discussion: - CV-4804 has an administrative limit of 2000 sccm. Three of the LLRTs (April 26, 2004, October 13, 2005, and November 2, 2008) were under this limit and hence did not require an apparent cause determination.

March 23, 2010 test: - CV-4804 failed its as-found LLRT on March 23, 2010. During the LLRT a packing leak was observed. Maintenance was performed on April 9, 2010. When

the valve was disassembled minor scratches were observed on the disc. A new cage, stem, and plug were installed, and the valve was repacked. An as-left LLRT was performed on April 14, 2010, with satisfactory results (29 sccm against an administrative limit of 2000 sccm). The apparent cause of CV-4804 leakage was the packing leak and scratches on the disc.

March 27, 2013 test: - CV-4804 failed its as-found LLRT on March 27, 2013. Maintenance was performed on June 17, 2013. A new cage, plug, stem and gaskets were installed. The apparent cause of CV-4804 was a damaged plug.

MU-36B discussion: MU-36B failed its as-found LLRT on October 23, 2011. Maintenance was performed on November 5, 2011. During the maintenance activities trash was observed in the valve seat. The valve internals were replaced. An as-left LLRT was performed on November 5, 2011, with satisfactory results (35 sccm against an administrative limit of 1500 sccm). The apparent cause of MU-36B leakage was trash on the valve seat.

HV-150 discussion: HV-150 has an administrative limit of 5000 sccm. Three of the LLRTs (October 12, 2005, April 5, 2010, and April 8, 2010) were under this limit and hence do not require an apparent cause determination. Another LLRT (May 1, 2004) results were 5000 sccm which did not exceed the administrative limit of 5000 sccm, hence, an apparent cause determination was not required.

October 11, 2002 test: - No detectable leakage was observed coming from the test vent during the October 11, 2002, LLRT. At the time of the test Steam Generator level was rising which means the observed leakage was through one or more drain valves that act as test boundaries. In addition when the outboard containment isolation valve was tested its leakage rate was 38 sccm. HV-150 serves as a boundary valve when the outboard containment isolation valve is tested. The apparent cause for HV-150 leakage was a leaking boundary valve not HV-150 leakage.

November 1, 2011 test: - The November 1, 2011, test was an as-left test following blue checking the disc and machined disc. Prior to this activity, HV-150 had passed its as-found LLRT which was performed on October 19, 2011 (88 sccm against an administrative limit of 5000 sccm).

On March 26, 2013, it was documented that HV-150 shaft was sheared. The sheared shaft precluded closing the valve and performing an as-found LLRT. During 2R25, the HV-150 shaft and packing were replaced. An as-left LLRT was performed on July 15, 2012, with satisfactory results (910 sccm against an administrative limit of 5000 sccm). The most likely cause of the November 1, 2011, as-left LLRT being above the administrative limit was debris under the seat.

CV-6205 discussion: This valve was cycled and LLRT tested following failure of the as-found LLRT (April 10, 2012 74800 sccm). After the valve was cycled its as-left leak rate

was 1790 sccm (administrative limit of 2000 sccm). The most likely cause for failing the as-found test was debris under the valve seat.

SV-1440 discussion: SV-1440 has been removed from the Appendix J program. The line that the valve is in has been cut and capped. This valve is not exposed to the containment atmosphere and serves no containment isolation function.

C-2 Escape Hatch Outer Door Seal discussion: C-2 failed its LLRT on November 16, 2011. The Escape Hatch was lubricated, and the outer door seal was shimmed to increase sealing pressure on November 17, 2011. The apparent cause was the outer door was not shimmed.

C-4 Personnel Hatch discussion: C-4 failed its as-left LLRT on July 31, 2013, with a leak rate of 447 sccm against an administrative limit of 200 sccm. This failure occurred at the end of the outage. Work has been initiated to correct C-4 leakage during the next refueling outage.

12. Please provide the following information for penetrations/components that are subject to Type Band Type C testing:

(a) Total number of penetrations/components subject to Type B test.

ANO-1 has 46 penetrations with 55 components that are Type B tested.

(b) Total number of penetrations/components subject to Type C test.

ANO-1 has 37 penetrations with 104 components that are Type C tested. Note three of these penetrations also have a Type B-tested component. These three Type B components are included in the 104 noted above. One LLRT may test more than one component. Blind flanges that are tested as part of a Type C test boundary are not counted.

(c) Total number of penetrations/components that are on an extended performance-based test interval compiled by their test interval (120-month, 50-month, 30-month, etc.)

Type B components are tested on a 120 month frequency with the following exceptions. The hatches (four penetrations / nine components) are tested on a 30 month frequency. The other exception are the three Type B tests listed in the response to 7 (b) above. They are tested on a 60-month frequency. The components (40 penetrations / 40 components) tested on a 120-month frequency are broken into groups, and one group is tested every refueling cycle. If one of the components that is on the 120-month (or 60-month) frequency fails an LLRT then it is placed on a 30-month frequency until it has passed two consecutive as-found LLRTs at which time it is returned to 120-month (or

60-month) test frequency. Currently no Type B-tested component that is on the extended test frequency (60 or 120-month) is on the accelerated 30-month test frequency.

Type C components are tested every 60-months with the following exceptions. The purge valves (two penetrations / four components) are tested on a 30-month frequency. Another exception is a component that has failed an LLRT. If a component has failed an LLRT it is placed on a 30 month test cycle until it has passed two consecutive as-found LLRTs at which time it is returned to a 60-month test frequency. A component may also be place on 30-month test frequency if an as-found test was not performed prior to maintenance activities that could affect its leakage characteristics or if a design change rendered prior leakage history irrelevant.

There have been eight LLRT test failures affecting seven components the last two outages. These components are therefore, on a 30-month test frequency. One component that failed its LLRT is located in a penetration that has been cut and capped and therefore, the component was not included in the 7(b) count. Two other components were placed on a 30-month test frequency because as-found testing was not performed prior to maintenance activities. One other component was placed on a 30-month test frequency because its spring was changed to a new design such that baseline testing needs to be established prior to returning it to a 60month test frequency. One of the components that failed also had its spring replaced. Another of the components that failed also did not have an as-found test performed. If the component failed its LLRT, it was not included in the other two categories (no as-found test or new design).

Response to RAI 2(a) and 2(b)				
Attribute	IE PRA Status	Discussion of F&O Basis	Basic Evaluation of PRA Importance	Impacts ILRT Evaluation
Method specific limitations	Open for Internal Events	Method-specific limitations and features that could impact results are not identified.	Quantifications were performed with computer codes that have been qualified under the Entergy Software Qualification Process. This finding is a documentation issue that has no impact on the PRA.	No impact on ILRT Evaluation. Results are based on industry standard codes and approaches.
		Identify and document the limitations and features of the methodology that could impact the PRA results.		
Plant specific data for internal flooding initiating event (IE) frequencies	Closed	There is no indication of use of plant specific operating experience or initiator information in the determination of IE frequencies.	A review of plant Operating Experience (OE) was performed by reviewing the Condition Report (CR) database for internal flooding related issues. This review was documented in a revision of the internal flood analysis. No plant related OE was found that would affect the generic frequencies used in the internal flood analysis.	No change based on review of operating experience so no change in current assessment and does not impact ILRT Evaluation.
		Obtain and integrate plant specific failure information into the calculation of the internal flooding initiating event frequencies.		
Internal flooding IE development	Open for Internal Events	There is insufficient information to determine that this Surveillance Requirement is met.	This finding is a documentation issue related to calculation of internal flooding IEs.	Documentation issue only so does not impact ILRT Evaluation
		Document the calculation process for internal flooding IEs more thoroughly, accounting for the SRs in ASME/ASN-RA-SA-2009, Section 2-2.1.		

Modeling as-built, as-operated plant	Open for Internal Events	PRA-A1-01-001S11, Rev 1., Section 4.0 documents the plant walkdowns and system engineer discussions for each system. The system engineer discussion is part of the latter. There is no indication in the walkdown/discussion documentation that the modeling was verified to represent the as-built, as-operated plant.	This finding is a documentation issue. The text of the finding acknowledges that the walkdowns and discussions were conducted, but that they were not sufficiently documented.	Documentation issue only so does not impact ILRT Evaluation
		Although it is acknowledged that system walkdowns and discussions with system engineers have been conducted and that spatial and environmental hazards were identified, the documentation of these activities does not convey that the model indeed does represent the as built/as-operated plant. Perform/document additional walkdowns and/or discussions focusing on confirmation that the model represents the as-built, as-operated plant, or document the satisfaction of this requirement if the existing walkdowns and/or discussions have already accomplished this goal.		

Phenomenological conditions associated with each accident sequence	Open for Internal Events	There is no purposeful description of the phenomenological conditions associated with each accident sequence, as required by the supporting requirement. Include a description of the phenomenological conditions for each sequence.	This finding remains open only as a documentation issue for the internal events model.	Documentation issue only so does not impact ILRT Evaluation
Internal flooding IE development	Closed	Unevaluated scenarios are grouped into similar but analyzed internal events, however, the impact of the flood may not be 'the same as the plant initiating event group already considered in the PRA.' Re-evaluate scenarios that were not considered further based on comparison of IE frequencies alone.	The Internal Flood Analysis has been revised to calculate the CDF and LERF for all scenarios that have been identified. The new revision to the analysis does not screen or subsume any scenarios or zones. The issue in this finding has been addressed in the revision to the internal flood analysis.	Update calculation addresses the F&O and the ILRT evaluation is not sensitive to this F&O.
Quantification of importance measures	Open for Internal Events	The quantification approach which includes modularization of the IE fault trees precludes calculation of importances for events within the modules. Provide discussion or tabulation of significant contributors to CDF from IEs as well as from mitigating systems.		Importance measures are not required for ILRT evaluation so the ILRT is not sensitive to this F&O.

Sources of uncertainty	Open for Internal Events	This uncertainty 'characterization' has not been performed. Perform a characterization of the sources of uncertainty. It is recommended that the EPRI 1016737/NUREG-1855 approach be applied.	A sensitivity and uncertainty analysis has been performed on the internal events model. The sensitivity analysis characterizes the sources of uncertainty which compare favorably with the issues identified in EPRI 1016737/NUREG-1855. The issue remains open only as a documentation issue.	Uncertainty measures are not required for ILRT evaluation so the ILRT is not sensitive to this F&O.

Basic event (BE) importance	Open for Internal Events	The guideline instructs that reviews include a comparison of the basic event risk importances and system importances in the current model to the previous model. This appears to address the BE review question but there is no explicit discussion of the BE review.	BE reviews were performed. This finding recommends that the BE reviews be tabulated to demonstrate the review.	Documentation issue only so does not impact ILRT Evaluation
		It would be helpful for the BE importances to be tabulated to demonstrate that the BEs had been reviewed.		
Internal flooding – flow rates/flood levels	Closed	Documentation of flood scenarios for the Aux. Building, Turbine Building, and Intake Structure are explained in sufficient detail and organized by subsection under Section 4.2 of PRA-A1-01-002. However, the calculational details of the reported water heights and flow rates reported for the analyzed scenarios were omitted from the documentation.	This issue was addressed in a revision of the Internal Flood Analysis.	Update calculation addresses the F&O and the ILRT evaluation is not sensitive to this F&O.
		Provide details of the calculated values that support the analyzed flood scenarios reported in Section 4.2 as either an appendix to PRA-A1-01-002 or as a reference to other standalone documents and calculations.		

LERF analysis limitations	Open for Internal Events	Reviewed PRA-A1-01-001S12, Revision 1. Section 2.1 identifies several limitations of the applicability. However, the noted limitations do not address technical limitations that might impact the use in applications.	Quantifications were performed with computer codes that have been qualified under the Entergy Software Qualification Process.	Update calculation addresses the F&O and the ILRT evaluation is not sensitive to this F&O.
		Document the limitations of the technical aspects of the analysis.		

Initiating events fault tree – event calculation method	Open for Internal Events	Some basic events (e.g., XMP119BBAF) applied in the calculation of IE frequencies developed for plant-specific fault trees have used calculation method 3 in CAFTA. The use of calculation method 3 (1-e-λt) produces a probability (always < 1) rather than a frequency which can be greater than 1. Calculation method 1 (λt) in CAFTA should be used for those basic events whose result is intended to be a frequency of failure, not a probability of failure.		The initiating event frequencies driving the significant internal event risk are less than 1.0 in frequency so this is not a concern for the ILRT Evaluation.
		A discussion of the use of this calculation method is not provided, although, during discussion of this issue, the PRA staff indicated that the limitations of the selected approach were understood.		
		Provide a description of the approach taken for calculation of the basic event values within support system initiating event fault trees, and include the limitations of the approach.		

IEs– critical years/reactor years	Open for Internal Events	Initiating event frequencies are not calculated in the manner required by the IE-C5. The IE units are in critical years versus reactor (calendar) years.		The inclusion of the annual period instead of the critical period would tend to adjust the results uniformly and the ILRT baseline results would be slightly altered but it would not impact the analysis delta change significantly.
		Calculate the frequencies in the units specified by SR IE-C5.		
		Section 5.3 of PRA-A1-01-001S06, Revision 2 identifies those initiating events that are quantified by means of a plant specific fault tree.		
		Appendices C, D, E, and F provide additional detail on each of the 4 modeled initiating events.		
		Per Appendices E and F, the PSA logic model is used as the starting point for the IE model; however, a number of modeling simplifications are made as identified in Appendix C. These simplifications may cause the model to fall out of compliance with the SY requirements.		
		Use the system fault tree with necessary data (exposure time) changes to evaluate IEs.		

HFE modeling for LERF	Open for Internal Events	Direct linkage of the CDF sequences into the LERF tree assures that system level dependencies (e.g., power and cooling water) can be accounted for.	The action to isolate the Steam Generator secondary side is not modeled so HRA and dependency analysis are not needed for this action.	Review of model for F&O resolution appears to support current modeling is accurate. Therefore no impact on ILRT evaluation. None of the identified issues would tend to increase the relative contributions to the different accident classes.
		No HRA for human actions is provided. Develop the quantification of human actions embedded in the LERF model using an HRA method consistent with the HRA SRs. The following actions have been noted to be present in the analysis:	HRA calculations consistent with the SRs have been developed for the HFEs to Depressurize by Opening pressurizer Power Operated Relief Valve and to bump the Reactor Coolant Pump. These calculations show that the existing values for these actions are appropriate.	
		- Depressurize by opening PZR PORV	The actions to Depressurize by Opening PZR PORV and Bump the RCP have a completely different purpose from the Level 1 HFEs, have relatively long time interval from the Level 1 HFEs, and are performed at the direction of the technical support center under the SAMGs (this means that cues are completely different from the Level 1 HFEs). Therefore, the dependency between them and the Level 1 HFEs is zero and dependency analyses are not needed.	
		- Isolate the SG secondary side		
		- “Bump” the RCPs		
		Accounting for dependency in human actions should be accounted for in the analysis. For example, some cutsets contain the following pair of events: NOT_RCS_DEP_NOSBO and RHF1HPIERP. Event NOT_RCS_DEP_NOSBO is in part a human error ‘Split Fraction for No Intentional or Unintentional		

		RCS Depress pre Induced-SGTR for Non-SBOs.[9]		
		Ideally, the human and hardware components of Split Fraction for No Intentional or Unintentional RCS Depress pre Induced-SGTR for Non-SBOs should be separated. A joint human error probability analysis can then be included in the result.		

Data – generic versus plant specific	Open for Internal Events	Reviewed PRA-A1-01-001S05_EC15022. Section 3.1.5 provides guidance on checking results for those components where no plant failures have occurred. If the check (operating hours > 0.5*MTTF) fails, the generic data is used. The intent is to guard against undue influence of not statistically significant plant data.	This finding identifies a discrepancy in the determination of particular plant-specific failure rates and distributions. A review of the data found five type codes (ACF L1, MFF E1, MPF L1, RYT P1 and T7F D1) for which the generic and plant-specific data were radically different. Corrected distributions were calculated. This issue impacts the uncertainty parameters with minimal impact on the point estimate value.	Results reflect updated model so no impact on ILRT evaluation.
		Guidance to check results is provided in CE-P-05.07, Rev. 01. Some posterior distributions do not appear reasonable (e.g., RYT P1, T7F D1).		
		Two examples of unreasonable distributions were found. The available guidance may not be sufficiently detailed to provide confidence that bad distributions are detected.		
		Provide additional examples in the guidance of what constitutes "unreasonable" to improve the confidence of detection and correction.		

Spurious Open of SRV	Closed	Reviewed cutset file A1_R4_1e-13_rec_M and fault tree A1R4P0EOS and PRA-A1-01-001, Revision 1, Section 6.4. This observation results from comparisons to similar plants.	Spurious opening of SRV is not classified as a %T2 (loss of feedwater) event.	The updated results used for the ILRT evaluation includes this change and there is no impact.
		Spurious opening of SRV evaluation needs to be reexamined.	A spurious opening and stuck open ERV is treated in initiating event %IORV, "Inadvertent Open Relief Valve." This event represents an inadvertent opening of the ERV which causes a reduction in primary system pressure, a reactor scram, and loss of RCS inventory. ANO-1 has three relief valves (one ERV and two code safeties) on the pressurizer. However, unlike the ERV which can be opened by a high pressure signal, the SRVs are mechanically operated by high RCS pressure. In addition, the plant will trip on high pressure prior to lifting the SRV. Also, the ERV setpoint is lower than the SRV and will lift prior to the RCS reaching the SRV setpoint. The IORV event is considered a small LOCA.	
		Revisit the classification of this event as %T2. Update the initiating event analysis with the proper classification of this event.		

LERF reasonableness review	Open for Internal Events	Sensitivity studies are performed for important inputs to the analysis. Reviewed PRA-A1-001-S02, Appendix J. No indication that the expert panel reviewed the LERF results. The SR requires a review of contributors for reasonableness. The expert panel report provides this for the CDF results but provides no indication that the LERF results were also reviewed. Include a review of the LERF contributors as part of the expert panel review. Document this review in the expert panel report.	This appears to be a documentation issue.	Documentation issue only so does not impact ILRT Evaluation. Also, the ILRT evaluation addresses only the best estimate results and is not directly influenced by any sensitivity studies.
Use of MAAP for LLOCA	Open for Internal Events	MAAP 4.0.5 provides detailed core damage sequences. PRA-A1-01-0015S014-EC14882 Section 4.0 A of the Level 1 Success Criteria Notebook credits the use of MAAP for a LLOCA blowdown phase. Based on current MAAP guidance, MAAP should not be used to model the blowdown and reflood stages of a LLOCA. DBA codes should be used in this case. Following reflood, MAAP can be used for the remainder of the LLOCA. Use a DBA code to determine LLOCA success criteria during the blowdown and reflood stage.	This finding identifies that the success criteria assume that only one CFT is required in order to flood the Reactor after a Large Break LOCA. The basis of this assumption is MAAP. However, MAAP is limited in its ability to model the blowdown phase of a large break LOCA and should not be used to justify success criteria.	Large LOCA is a small contributor to core damage and would not be significantly impacted by a change to require two core flood tanks. The ILRT Evaluation is not impacted by this F&O.

Modeling uncertainty	Open for Internal Events	There is no discussion of identification of issues related to modeling uncertainty. Provide the identification of sources of modeling uncertainty. It is recommended that the process described in EPRI 1016737/NUREG-1855 be incorporated.	A sensitivity and uncertainty analysis has been performed on the internal events model. The sensitivity analysis characterizes the sources of uncertainty which compare favorably with the issues identified in EPRI 1016737/NUREG-1855. The issue remains open only as a documentation issue.	Documentation issue only so does not impact ILRT Evaluation
Data – counting operational demands	Open for Internal Events	There is no evidence that surveillance tests have been evaluated to determine if portions of the tests or sub-elements have additional successes that should or should not be counted when estimating operational demands. Perform an assessment of the sub-elements of all surveillance tests to obtain accurate operational demands to be used in the PRA data.	The consideration of demands to determine the probability of failure is limited to those failure rates calculated from plant specific data analysis. This fact limits the number of Type Codes that must be visited to address this F&O. Also, since demands on the major component being tested are counted, additional operational demands within surveillance sub-elements would affect only supporting components, further limiting the affected Type Codes.	Underestimating successful operations would result in conservative failure data which would overestimate CDF and LERF. Therefore, the increases presented in the ILRT Evaluation are possibly conservative with respect to the resolution of this F&O.

Data – counting failures	Open for Internal Events	The primary source of failure data is the Maintenance Rule Database. This database was used to screen the component failures to determine if the Maintenance Rule Functional Failures were also PRA-relevant failures. It is suggested that an additional source of data be reviewed to determine if a failure may have occurred that did not result in a Maintenance Rule Functional Failure. In addition, a suggestion is provided to include a discussion or tabular display of those failures that are excluded from the data.	The consideration of demands to determine the probability of failure is limited to those failure rates calculated from plant specific data analysis. This fact limits the number of Type Codes that must be visited to address this F&O.	Typically failures are well captured by the PRA update process and the main issue would be associated with counting demands and operational hours. There is some potential for failure events to be uncovered but the impact would be small on the individual component failure data and expected to be minimal on the overall CDF. The impact on the ILRT Evaluation would be very limited.
		Perform a scrub or review of EPIX, Condition Reports, Issue Reports, and/or plant specific LERs to determine if there are any additional failures that should be considered in the PRA Data Update to supplement the Maintenance Rule Database functional failures.	Potential changes in the random failure probabilities from resolution of this finding would have a very small impact on the PRA results.	

Alternate AC (AAC) alignment	Open for Internal Events	Interviews with knowledgeable plant personnel are documented in various locations in the PRA Data Analysis in various assumptions and as sources of data to estimate run times, demands, etc. It is not clear that Outage UA has been excluded from the Maintenance rule Data since some Maintenance rule functions may be outage related. There is no consideration for alignment of the AAC during a dual unit SBO. In addition it appears that the SU2 transformer could be credited to support both trains on Unit 2, however it is assumed to be aligned to Unit 1.	The issue of use of outage unavailability data is a documentation issue. This issue has no impact on the PRA.	The identified F&O is at most a documentation issue and does not impact the ILRT Evaluation.
			The effect of AACDG unavailability due to dual unit SBO is negligible compared to AACDG unavailability due to test and maintenance. The potential for a LOSP on both units with failure of four emergency diesel generators requiring the shared use of the AACDG is of such low probability that it does not affect results. In addition, the AACDG is sized such that it can carry some critical loads on each unit if required.	
		Model the AAC unavailability to support either unit in the event that both or either unit requires it for LOSP mitigation and provide a documented basis for the flag alignment settings used for the SU2 transformer.	The model does not assume that SU2 is aligned to Unit 1. Logic below gate ELSSU2, “SU2 ALIGNMENT BETWEEN UNITS OVERLOAD” accounts for the fact that SU2 is normally aligned to be shared between the two units. This finding has no impact on the PRA.	

Total loss of Service Water (SW)	Closed	Section 4.0 of PRA-A1-01-001S06, Revision 2 describes the process used to identify initiating events. This process considered generic events as well as initiating events modeled in similar plants (TMI and Oconee). It was noted that the loss of Service Water (SW) initiator is relatively low significance in the ANO-1 PRA model. %T8, %T9, and %T10 are identified as Loss of Running SW, Loss of SW Loop 1 and Loss of SW Loop 2 respectively. There are no initiators that represent a total loss of all 3 SW Pumps (including common cause of all three to fail). An initiator for Loss of Lake is included in the model but that does not account for SW pump failures. In similar NSSS designs, the loss of SW is a significant contributor to CDF. The process is further prescribed in Fleet engineering guide EN-NE-G-006.	Initiator %T8, “Loss of Running SW” addresses total loss of all three SW pumps, including common cause failure of all three pumps and common cause failure of all three discharge filters. This initiator also considers loss of two pumps and failure of the cross-tie valves to close, since one pump cannot supply both loops by itself.	Based on inclusion of %T8, the item is closed and the ILRT Evaluation is not impacted.
		Include the total loss of SW in the ANO-1 PRA.		

SGTR modeling	Open for Internal Events	The development of the Event Trees appears to be consistent with the plant design/operation. An isolated model error was found related to placement of a Human Action to cooldown and depressurize during a Steam Generator Tube Rupture. The error results in no account for HEP probability to fail to initiate the cooldown process high enough up in the SGTR event tree. An HEP should be place near the top gate to yield simple sequences where an SGTR occurs, no equipment failures occur but the operators fail to cooldown and depressurize.		The SGTR sequences are considered LERF and are associated with Class 8. Any increase in CDF would be carried through all cases and would have the effect of reducing the calculated change in risk presented in the ILRT evaluation and the current assessment would be conservative.
		Add an HEP to the SGTR sequence model high enough in the model logic to verify that OPS successfully initiates the SGTR cooldown.		

IEs – lack of justification for exclusion	Open for Internal Events	In Table 3 of PA-A1-001S06, Rev. 2, where screened potential initiating events are considered, two human recovery actions are used to justify not modeling an event as an initiator (i.e., Steam line break and High Pressure Injection (HPI) actuation). However, no justification (i.e., training or procedures) was provided.	This is considered to be a documentation issue.	Documentation issue only so does not impact ILRT Evaluation
		Provide the appropriate training documents or procedures showing these particular human actions.		
Spurious HPI	Open for Internal Events	The screening performed in Table 3 of PRA-A1-01-001S06, Rev. 2 generally follows the conditions specified in SR IE-C6. However, the conditions in this SR are not explicitly involved, i.e., the 10E-8 frequency was used with an order-of-magnitude argument to discount the initiator, not criteria (a).		The inclusion of a spurious injection event could slightly increase the baseline CDF but would be a proportional increase across the various classes such that the overall impact would be minimal. It would not be expected to significantly impact the ILRT evaluation.
		As the ASME/ANS PRA Standard is currently written, the screening criteria in IE-C6 need to be used. The calculation for spurious HPI actuation needs to be checked.		
		Add spurious HPI actuation due to spurious ESAS actuation as an additional initiating event.		

Plant specific pre-initiators	Open for Internal Events	The ANO-1 PRA Peer Review road map indicated that the pre-initiator events have been reviewed against plant-specific failures. To be assessed at CC II/III for this SR, a list of existing pre-initiator events at ANO-1 needs to be prepared, and it needs to be compared to the list in Table 2. Events not appearing in Table 2 would need to be added.	An extensive number of pre-initiator HRAs are modeled. These events cover all standby systems and trains. Thus, this finding is just a documentation issue. To obtain CC II/III classification, more thorough documentation of the plant specific input into the pre-initiators could be provided.	Documentation issue only so does not impact ILRT Evaluation
Success Criteria – comparison to other plants	Open for Internal Events	No documentation exists that describes comparisons with similar plants or other plant specific codes to check the reasonableness and acceptability of the results of the thermal/hydraulic, structural, or other supporting engineering bases that support the success criteria. Perform a comparison with other plants and document.	This finding indicates that the documentation of the success criteria should include a comparison with similar plants.	Documentation issue only so does not impact ILRT Evaluation

ISLOCA modeling	This F&O has been resolved in association with the FPRA.	There was no reference made to surveillance tests at power, if applicable, in which the ISLOCA pathway configuration would be changed from its routine configuration and alignment, e.g., two isolation valves instead of three. Common cause mechanisms were discussed as not being applicable when they should have been included.	The FPRA model ISLOCA logic was revised to capture the potential for a human performance error in the restoration of the DHR isolation valve CV-1400/1401. LHF1LPITNA was added to the ISLOCA logic to address this issue.	The updated results used for the ILRT evaluation includes this change and there is no impact.
		Reference testing procedures and their frequency in order to more accurately account for the time when the ISLOCA pathway is in a different configuration (i.e., two valve isolation instead of three). Consider common cause failure mechanisms, which also are related to 'state-of-knowledge' correlation (see QU-A3).	Appendix B of NUREG/CR-5102, “Interfacing Systems LOCA: Pressurized Water Reactors,” evaluates the probability of multiple failures and concludes that a CCF event for check valve leakage or rupture is not needed.	
			Also, %FRDH14ASOK, %FRDH14BSOK, %FRDH17SOK, and %FRDH18SOK were added to the FPRA ISLOCA logic to capture shared failure of check valves due to state of knowledge correlation.	

Conservative treatment of secondary piping	Open for Internal Events	The capability of secondary system piping appears to be such that once the isolation valves fail, the low pressure piping automatically fails.	This finding identifies a conservative assumption that was not documented. Thus, it has no impact on the PRA.	Documentation issue only so does not impact ILRT Evaluation
		Since it appears that there was no consideration given to secondary piping capacity, e.g., fragility analysis, this could be a conservative treatment.		
		State that a conservative approach was taken by assuming automatic failure of secondary piping once it is exposed to high pressure, either via leak or rupture.		
Internal flooding – plant specific maintenance	Open for Internal Events	The EPRI failure database in TR-1013141 excluded certain events in the calculation of pipe failure frequencies that appear to be related to maintenance activities. See Table C-2, e.g., Crystal River 3 event.		The updated results used for the ILRT evaluation includes this change and there is no impact.
		Devise a method or process in which the contribution to internal flooding due to plant-specific maintenance activities is estimated and incorporated into the various existing internal flood initiators.		

Common cause modeling	Open for Internal Events	In reviewing the actual fault tree model, it was seen that the CCF terms were incorporated per the system modeling documentation (Supplement 11) and CCF documentation (Supplement 4). However, there was an inconsistency with what was stated in Supplement 4 and what was incorporated in the PRA model. Please resolve discrepancy between what was recommended in the common cause calculation and what was done in the PRA model.	This finding identifies a documentation issue.	Documentation issue only so does not impact ILRT Evaluation
Internal flooding – initiator screening	Closed	In reviewing PRA-A1-01-002, it was noted for several scenario frequencies that they were screened on a strict comparison with the internal events initiating frequency, which does not comport with this particular SR. A few examples may be found in Sections 4.2.1.50, 4.2.1.52, and 4.2.1.36. Re-evaluate those scenario groups that were screened from further evaluation based solely on a comparison of initiating frequencies alone.	This issue has been addressed in a revision to the internal flood analysis. A revision of the scenario write-up and quantification of all scenarios in which a flood frequency exists was performed.	The updated results used for the ILRT evaluation includes this change and there is no impact.

Internal flooding – flood damage classification	Open for Internal Events	Upon review of the internal flood document (PRA-A1-01-002) and other supplemental files, it was not explicitly clear as to what specific SSCs within a given flood area were susceptible to a particular flood damage category, e.g., submergence or spray. Provide a listing of the SSCs organized by flood zone and associated scenario with a listing of their susceptibility to flood damage due to both spray and submergence. For RG 1.200, a qualitative assessment involving conservative assumptions regarding the additional flood damage mechanisms under Capability Category III should be considered.	A revision to the Internal Flood Analysis was performed to address the issue identified in this F&O. Information was included in Section 6.4 that provides information relating to the affected SSCs modeled in the PRA, the assumptions regarding spatial information, and flooding affects assumed in the analysis. This risk calculations performed in this revision included all PRA modeled components affected by a flood in each of the zones.	The updated results used for the ILRT evaluation includes this change and there is no impact.

ISLOCA – state of knowledge correlation	Open for Internal Events	The state of knowledge correlation was not accounted for where it would make a significant difference, i.e., the ISLOCA analysis omitted common cause failure of check valves. Document PRA-A1-01-001S08 was reviewed to confirm this.	This finding remains open for the internal events model.	The inclusion of the state-of-knowledge correlation will tend to increase the ISLOCA contribution which will increase both the baseline value and the extension value the exact same. This would be classified as Class 8 increases and would tend to reduce the delta changes associated with the ILRT extension and the current results would be marginally conservative.
		Consider common cause failure mechanisms, which also may be related to 'state-of-knowledge' correlation (see QU-A3).	Appendix B of NUREG/CR-5102, “Interfacing Systems LOCA: Pressurized Water Reactors,” evaluates the probability of multiple failures and concludes that a CCF event for check valve leakage or rupture is not needed.	
			Also, %FRDH14ASOK, %FRDH14BSOK, %FRDH17SOK, and %FRDH18SOK were added to the FPRA ISLOCA logic to capture shared failure of check valves due to state of knowledge correlation. This finding has been resolved in the FPRA.	

Quantification limitations	Open for Internal Events	A review of the Integration and Quantification Work Package and the FORTE Qualification Engineering Report did not reveal any documented software or quantification limitations that would impact applications. Document any known quantification limitations, and if none, state that there are no known limitations.	Quantifications were performed with computer codes that have been qualified under the Entergy Software Qualification Process. This finding is a documentation issue that has no impact on the PRA.	No impact on ILRT Evaluation. Results are based on industry standard codes and approaches.
Documentation of risk significant items	Open for Internal Events	Attachment E of the Summary Report was found to list significant accident sequences and basic events. Attachment C lists the top 25 cutsets. Provide definitions for risk significant basic events, cutsets, and accident sequences within the Summary Report that comport with those listed in Section 1-2.2 of the ASME Standard.	This finding identifies a documentation issue related to the reporting of risk significances in the summary report and has no impact on the PRA results.	Documentation issue only so does not impact ILRT Evaluation

Internal flooding documentation	Closed	Although flood sources were documented and discussed within Section 4.2 of PRA-A1-01-002, they are not amenable to PRA applications and upgrades. Supplemental Excel spreadsheets were obtained and flood sources were listed by flood zone, but it was confusing as to why different lengths were used for general and major flood scenarios. Also, there was a lack of clarifying information as to why certain pipe lengths that were considered for flood scenarios were excluded from spray scenarios.	The internal flooding analysis has been updated to address this finding.	The updated results used for the ILRT evaluation includes this change and there is no impact.
		One method of satisfying this SR would be to provide a tabular listing of water sources organized by flood zone, associated system, pipe diameter, pipe length, and corresponding flood scenario.		

Internal flooding documentation	Closed	In reviewing the flooding frequencies reported in Section 4.2 of PRA-A1-01-002 for each of the internal flood scenarios, it was not readily apparent how the frequencies were derived. Supplemental Excel spreadsheets were obtained that helped explain how some of these frequencies were derived, but they were not part of the formal documentation. One method of satisfying this SR would be to provide a table of the identified water sources that associates the calculated flood frequencies with each of the postulated internal flood scenarios.	The internal flooding analysis has been updated to address this finding.	The updated results used for the ILRT evaluation includes this change and there is no impact.
Internal flooding – flood related HEPs	Closed	The documentation in Section 4.2 of PRA-A1-01-002 takes credit for operator actions, e.g., operation of locked valves within a 15 minute time period, without incorporating a corresponding HEP event representing this action in the PRA model. Perform an evaluation of HEPs that were described in Section 4.2 of PRA-A1-01-002, e.g., via the use of the HRA Toolbox, or alternatively, apply temporary screening values as a sensitivity analysis.	The internal flooding analysis has been updated to address this finding.	The updated results used for the ILRT evaluation includes this change and there is no impact.

Internal flooding – door failure	Open for Internal Events	Although not necessarily considered an SSC, a review of PRA-A1-01-002 did not reveal any additional analysis regarding the water height at which a typical fire door would be considered to fail. Either reference or include within the internal flood report an analysis or relevant assumptions regarding door failure as a function of water height.	This is a documentation issue only and does not affect results or conclusions.	Documentation issue only so does not impact ILRT Evaluation
Internal flooding documentation	Open for Internal Events	Although some source capacities may have been mentioned in Section 4.2 of PRA-A1-01-002 for the description of the internal flood scenarios, there was minimal information regarding internal pressure and temperature of water sources. Provide a table within Section 4 of PRA-A1-01-002 that lists the various water sources considered and their corresponding system capacity, temperature, and pressure.	This is a documentation issue only and does not affect results or conclusions.	Documentation issue only so does not impact ILRT Evaluation