Enclosure A

Responses to Request for Additional Information and Request for Confirmation of Information (Non-Proprietary) Clinton Power Station, Unit 1 License Renewal Application (LRA)

RAI B.2.1.42-1
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Regulatory Basis

10 CFR 54.21(a)(3) requires an applicant to demonstrate that the effects of aging for structures and components will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation. One of the findings that the staff must make to issue a renewed license (10 CFR 54.29(a)) is that actions have been identified and have been or will be taken with respect to managing the effects of aging during the period of extended operation on the functionality of structures and components that have been identified to require review under 10 CFR 54.21, such that there is reasonable assurance that the activities authorized by the renewed license will continue to be conducted in accordance with the current licensing basis. In order to complete its review and enable making a finding under 10 CFR 54.29(a), the staff requires additional information in regard to the matters described below.

Background

GALL-LR Report AMP XI.M42, "Internal Coatings/Linings for In-Scope Piping, Piping Components, Heat Exchangers, and Tanks," as added by LR-ISG-2013-01, "Aging Management of Loss of Coating or Lining Integrity for Internal Coatings/Linings on In-Scope Piping, Piping Components, Heat Exchangers, and Tanks," recommends conducting baseline inspections in the 10-year period prior to the period of extended operation "in order to establish the condition of coatings/lining prior to entering the period of extended operation. In addition, these baseline inspections provide input to the interval of subsequent inspections."

LRA Section B.2.1.42, "Internal Coatings/Linings for In-Scope Piping, Piping Components, Heat Exchangers, and Tanks Program," describes operating experience from the Division 2 diesel generator fuel oil storage tank cleaning in 2015 in which fourteen separate indications related to the integrity of the tank internal coatings system were documented. LRA Section B.2.1.42 also states that an evaluation was performed which "determined that because the affected areas were very small, the remaining coated surface was in good condition, and no loose coating was identified that could affect downstream components, the tank could be returned to service until the next scheduled inspection. The next tank cleaning will be performed in 2025 with any coating repairs taking place during this time."

Issue

- It is unclear to the staff if "the next scheduled inspection" of the Division 2 diesel generator fuel oil storage tank, described in LRA Section B.2.1.42 under the operating experience description, will be in 2025 when the tank is next scheduled to be cleaned.
- It is also unclear to the staff if the next scheduled inspection of this tank will be considered as part of the baseline inspections recommended by GALL-LR Report AMP XI.M42.
- It is unclear to the staff if all fourteen indications in the Division 2 diesel generator fuel oil storage tank internal coating/lining, documented during a 2015 inspection, will be repaired during the tank cleaning proposed to be performed in 2025. If not all fourteen indications in the Division 2 diesel generator fuel oil storage tank internal coating/lining will be repaired during the tank cleaning proposed to be performed in 2025, it is unclear to the staff what acceptance criteria and corrective actions will be applied to the indications to assure

acceptability of the internal coating/lining as the tank enters the period of extended operation.

Request

1. Confirm the year of the next scheduled inspection of the Division 2 diesel generator fuel oil storage tank.
2. Clarify if the next scheduled inspection of the Division 2 diesel generator fuel oil storage tank will be considered as part of the baseline inspections recommended by GALL-LR Report AMP XI.M42.
3. Clarify if all fourteen indications in the Division 2 diesel generator fuel oil storage tank internal coating/lining, documented during a 2015 inspection, will be repaired during the tank cleaning proposed to be performed in 2025. If not all fourteen indications will be repaired, clarify what acceptance criteria and corrective actions will be applied to the indications to assure acceptability of the internal coating/lining as the tank enters the period of extended operation.

Constellation Response:

1. The next scheduled inspection will be performed in 2025.

2. Yes, the 2025 inspection will be the baseline inspection. The last inspection was performed in 2015. The "Internal Coatings/Linings for In-Scope Piping, Piping Components, Heat Exchangers, and Tanks," Aging Management Program (AMP) requires the baseline inspection to be performed within the 10-year period prior to entering the period of extended operation (PEO) which, for Clinton Power Station, will be in 2027. Therefore, the last inspection cannot be considered the baseline inspection because it was performed prior to entering the 10-year period before entering the PEO.

3. All 14 indications are scheduled for repair during the upcoming (2025) inspection.

RAI 3.1.2.2.1-1

Regulatory Basis

Pursuant to 10 CFR 54.21(c), the LRA must include an evaluation of time-limited aging analyses (TLAAs). The applicant must demonstrate that (i) the analyses remain valid for the period of extended operation, (ii) the analyses have been projected to the end of the period of extended operation, or (iii) the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

Background

LRA Table 3.1.2-3 addresses the aging management review (AMR) results for the jet pump assembly holddown beam in relation to AMR item 3.1.1-3 (LRA page 3.1-69). LRA Table 3.1.2-3 also addresses the AMR results for the jet pump assembly inlet riser, brace and sleeve, elbow, wedge, diffuser, and holddown beam bolts in relation to AMR item 3.1.1-3 (LRA page 3.1-70). The LRA table indicates that these jet pump components are subject to a fatigue TLAA.

Issue

LRA Section 4.3.7.1 addresses the fatigue TLAA for the jet pump riser brace. However, the LRA does not clearly describe the fatigue TLAA evaluations and dispositions for the other jet pump components discussed in the background section above (i.e., jet pump assembly holddown beam, inlet riser, sleeve, elbow, wedge, diffuser, and holddown beam bolts).

Therefore, the staff needs to clarify the following items: (1) which specific jet pump components discussed in the background section are subject to a fatigue TLAA; (2) specific LRA sections that describe the fatigue TLAA evaluations and dispositions for the jet pump components subject to a fatigue TLAA; and (3) whether the jet pump riser brace is bounding for the other jet pump components in terms of fatigue analysis (e.g., in terms of cumulative usage factor (CUF) analysis).

Request

1. Describe which specific jet pump components discussed in the background section are subject to a fatigue TLAA.
2. Clarify the specific LRA sections that describe the fatigue TLAA evaluations and dispositions for the jet pump components subject to a fatigue TLAA.
3. Clarify whether the jet pump riser brace is bounding for the other jet pump components in terms of fatigue analysis (e.g., in terms of CUF analysis). If so, discuss how the applicant determined the bounding nature of the jet pump riser brace.
4. Revise the LRA as needed to provide the fatigue TLAA evaluations and dispositions for the jet pump components, consistent with the discussion above.

Constellation Response:

1. The jet pump riser brace is the only jet pump subcomponent discussed in the background section that is associated with a fatigue TLAA as described and evaluated in LRA Section 4.3.7.1.

The jet pump instrumentation penetration seals fatigue TLAA is described and evaluated in LRA Section 4.3.4. However, this component is not listed in the background section of this RAI.

During the development of LRA Chapter 4, “Time-Limited Aging Analyses,” extensive searches were performed of the Clinton current licensing basis (CLB) as described in LRA Section 4.1. This resulted in the evaluation of approximately 7,000 documents including all sections of the USAR. The extensive searches showed that only the jet pump riser brace subcomponent and the jet pump instrumentation penetration seals component were evaluated for fatigue as they met the six criterion in 10 CFR 54.3. The extensive searches identified no other in-scope jet pump subcomponent fatigue evaluations as a part of the Clinton CLB.

LRA Table 3.1.2-3 Note 1 for the “Jet Pump Assemblies: Inlet riser, brace and sleeve, elbow, wedge, diffuser, holddown beam bolt” AMR line was imprecise. This note may lead to the conclusion that all or some of the subcomponents in this AMR line are associated with fatigue TLAAs. The designation of an associated fatigue TLAA for this AMR line was intended to reference a fatigue TLAA associated with only the jet pump riser brace subcomponent and not all the subcomponents listed in the AMR line.

LRA Table 3.1.2-3 Note 1 should have been more specific in that only the jet pump riser brace subcomponent in this AMR line is associated with a fatigue TLAA as evaluated in LRA Section 4.3.7.1. The remaining subcomponents addressed in this AMR line are not associated with fatigue related TLAAs.

LRA Table 3.1.2-3 Note 1 was revised in LRA Supplement 2, dated December 20, 2024, via Change #4. This supplement replaced Note 1 in the “Jet Pump Assemblies: Inlet riser, brace and sleeve, elbow, wedge, diffuser, holddown beam bolt” AMR line in LRA Table 3.1.2-3 with Note 5. Note 5 explains that the TLAA designation in the Aging Management Program column indicates that fatigue of the jet pump riser brace is evaluated in Section 4.3.

2. The jet pump riser brace is the only jet pump subcomponent discussed in the background section that is associated with a fatigue TLAA as described and evaluated in LRA Section 4.3.7.1. The jet pump instrumentation penetration seals fatigue TLAA is described and evaluated in LRA Section 4.3.4. No other jet pump subcomponents are associated with fatigue TLAAs.

3. The NRC approved a 20 percent extended power uprate (EPU) via Amendment No. 149, dated April 5, 2002, which authorized an increase in the maximum licensed thermal power level from 2894 MWt to 3473 MWt. In support of the EPU License Amendment Request (ML011720516), General Electric (GE) issued NEDC-32989P, “Safety Analysis Report for Clinton Power Station Extended Power Uprate (Proprietary),” which included an evaluation of impacted reactor vessel internal (RVI) components for fatigue using EPU operating conditions. The jet pumps were addressed in this report and the jet pump riser brace subcomponent was identified as the key jet pump subcomponent (e.g., bounding) with respect to jet pump fatigue with a design CUF value of 0.976. Therefore, the jet pump riser brace component fatigue will be monitored and managed by the Fatigue Monitoring (B.3.1.1) program through the period of extended operation as described in LRA Section 4.3.7.1.

4. No updates to the LRA are required as a result of these responses.

RAI 4.3.2-1

Regulatory Basis

Pursuant to 10 CFR 54.21(c), the LRA must include an evaluation of time-limited aging analyses (TLAAs). The applicant must demonstrate that (i) the analyses remain valid for the period of extended operation, (ii) the analyses have been projected to the end of the period of extended operation, or (iii) the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

Background

LRA Section 4.3.2 indicates that the screening values of environmental fatigue correction factor (F_{en}) are based on the component material, maximum operating temperature, and bounding dissolved oxygen. The LRA also indicates that sulfur content is also an input for the screening F_{en} values of carbon and low alloy steel components.

Issue

LRA Section 4.3.2 does not clearly discuss how the applicant determined conservative sulfur content (for carbon and low alloy steels) and strain rates in the screening evaluation for EAF. The staff also noted that the applicant reduced the conservatism associated with the screening environmentally adjusted cumulative usage factor (CUF_{en}) values in the detailed EAF evaluation after the screening evaluation. Therefore, the staff needs clarification on how the applicant reduced the conservatism associated with the screening CUF_{en} values in the detailed EAF evaluation.

Request

1. Describe how the applicant determined conservative sulfur content (for carbon and low alloy steels) and strain rates in the screening EAF evaluation.
2. Describe how the applicant reduced the conservatism associated with the screening CUF_{en} values in the detailed EAF evaluation after the screening EAF evaluation.

Constellation Response:

1. For the EAF screening evaluation, Constellation selected sulfur content values and strain rate values that would result in the maximum contribution to the screening Fen values ($F_{en60scr}$).

   In NUREG/CR-6909, Revision 1 the calculated Fen value for a given component is calculated using four major inputs. These are: strain rate, sulfur content, temperature, and historical dissolved oxygen concentrations. For example, NUREG/CR-6909, Revision 1 recommends that Fen values for both carbon and low-alloy steels are calculated as follows:

$$F_{en} = \exp((0.003 - 0.031\dot{\varepsilon}^*)\ S^* T^* O^*),$$

Where "$\dot{\varepsilon}^*$" is strain rate, "S" is sulfur content, "T" is operating temperature, and "O" is dissolved oxygen.

Therefore, the EAF screening evaluation used the most conservative sulfur content value recommended in NUREG/CR-6909, Revision 1. For example, for both carbon and low-alloy steels NUREG/CR-6909, Revision 1 recommends the following:

$S^* = 2.0 + 98\ S$ ($S \leq 0.015$ wt. percent)

OR

$S^* = 3.47$ ($S > 0.015$ wt. percent)

For the EAF screening evaluation, the value of 3.47 was selected for both carbon and low-alloy steels, since it results in the maximum contribution to Fen.

In addition, the EAF screening evaluation selected strain rates with the greatest contribution to Fen. For example, for both carbon and low-alloy steels NUREG/CR-6909, Revision 1 recommends the following:

$\dot{\varepsilon}^* = 0$ for ($\dot{\varepsilon} > 2.2$ percent/s)

OR

$\dot{\varepsilon}^* = \ln(\dot{\varepsilon}/2.2)$ for (0.0004 percent/s $\leq \dot{\varepsilon} \leq 2.2$ percent/s)

OR

$\dot{\varepsilon}^* = \ln(0.0004/2.2)$ for ($\dot{\varepsilon} < 0.0004$ percent/s)

Therefore, for the EAF screening evaluation, the value of "ln(0.0004/2.2)" was selected for both carbon and low-alloy steels, since it results in the maximum contribution to Fen.

2. The EAF screening evaluation assumed very conservative factors, such as the number of assumed transient occurrences and conservative operating temperatures, which resulted in significantly overly conservative screening Fen and screening CUFen values. The purpose of the evaluation was to conservatively identify bounding components and screen out components which are not bounding.

   The detailed evaluations removed unnecessary conservatisms from components that screened in from the screening evaluation. The detailed evaluations were performed per the guidance in NUREG/CR-6909, Revision 1 and accurately calculated projected CUFen values for the bounding components; consistent with the original fatigue evaluation and adjusted for Extended Power Uprate (EPU) and EAF. The resulting calculations are then incorporated into the FatiguePro™ software, if required.

   The "screening 60-year CUFen" values ($U_{en60scr}$) calculated in the EAF screening evaluation are more conservative than the "projected 60-year CUFen" values developed in the more detailed analyses. Described below are three of the more significant factors which support this conclusion.

1) The screening 60-year CUFen values were scaled from the design 40-year CUF by a factor 1.5 (60/40). The design 40-year CUF values for these locations is based on the assumed transients and an assumed number of transient occurrences, documented in LRA Table 4.3.1-1. LRA Section 4.3.1 and Table 4.3.1-1 show that the projected number of occurrences for 60 years are less than the number of occurrences originally assumed for 40 years, with three minor exceptions which are discussed in the response to RAI 4.3.1-1. Since usage values (CUF) are the ratio of the number of assumed occurrences (n) divided by the number of allowable occurrences (N), multiplying the design 40-year CUF by a factor of 1.5 effectively increases the number of assumed occurrences by 150 percent. Therefore, “screening 60-year CUFen” values have an inherent margin of at least approximately 50 percent than what is expected based on the 60 year transient occurrence projections documented in LRA Table 4.3.1-1.

   In contrast, the calculation of the "projected 60-year CUFen" values in the detailed evaluation assumed the projected 60-year number of occurrences based on Thermal Fatigue Monitoring program (B.3.1.1) cycle occurrence data.

   Therefore, the “screening 60-year CUFen” values are more conservative than the “projected 60-year CUFen” values since all the projected 60-year transient cycle occurrences in column 4 of Table 4.3.1-1 are significantly less than 150 percent of the number of assumed 40-year transient occurrences in the fifth column of LRA Table 4.3.1-1. Note the actual number of occurrences and the 60 year projections in LRA Table 4.3.1-1 are based on Thermal Fatigue Monitoring program (B.3.1.1) cycle occurrence data up to September 2022.

2) For the screening evaluation the calculation of the "screening 60-year CUFen" values assumed "bounding screening Fen" values based on the maximum specified temperature from the reactor cycle diagram for all the specified transients. In contrast, for the detailed evaluations the calculation of the "projected 60-year CUFen" values in LRA Table 4.3.1-2 developed individual Fen values for each specified transient. In cases where the specified transient is relatively simple or when thermal gradient stresses are dominant, an average transient temperature value from the specified transient profile was used. The average temperature of each specified transient is generally less than the maximum specified temperature from the reactor cycle diagram for all the specified transients. This is conservative since Fen multipliers increase exponentially with temperature. Therefore, the contribution due to the assumed maximum temperature in the screening evaluation is more conservative than the contribution of the average temperature of each transient used to calculate the "projected 60-year CUFen" values in the detail evaluation.

3) The screening evaluation multiplied the 60 year usage values (CUFs) by very conservative Fen values. Since usage values (CUF) are the ratio of the number of assumed occurrences (n) divided by the number of allowable occurrences (N), the multiplication by a conservative Fen value based on the maximum operating temperature effectively assumes that all the specified transient profiles are assumed to be equal to the maximum specified temperature from the thermal cycle diagrams. However, the ASME Section III and NUREG/CR-6909, Revision 1 methodology allows for the calculation of transient specific CUFen values for

each transient or load pair, based on the number of specified transient occurrences for the individual transient and a Fen value corresponding to the specified temperature profile for that transient, which is less than the maximum specified temperature from the reactor cycle diagram for all the specified transients. Then all the calculated CUFen values for each transient are added together into a total CUFen value for the component. In addition, the ASME Section III and NUREG/CR-6909, Revision 1 methodology allows for the pairing of transients into load pairs, to calculate usage values. This methodology allows for finer granularity in calculating a total CUFen.

This difference is best explained by an example, as follows. The screening evaluation concluded that the "node 60 recirc suction pipe" component (which is in region B) was assessed a $F_{en60scr}$ value of 9.348 based on the maximum specified temperature of 567 degrees F from the entire reactor cycle diagram. The resulting screening CUFen value was 8.787.

For this same component the detailed evaluation used 26 transient load pairs with average temperatures (ranging from 370.25 to 488.25 degrees F) established for each transient load pair; based on the original individual transient profiles specified in the reactor cycle diagram. Based on these average temperatures, individual Fen values were calculated for each of the 26 transient load pairs. Based on these individual Fen values, 26 different CUFen values were calculated for each load pair and the 26 CUFen values were tallied into a total 60 year projected CUFen value of 0.528. The calculated average Fen value of the 26 load pairs is approximately 2.932. The refined CUFen and Fen values based on the methodology in NUREG/CR-6909, Revision 1 are more accurate.

These three factors result in the removal of unnecessary conservatism from the screening evaluation.

RAI 4.3.2-2

Regulatory Basis

Pursuant to 10 CFR 54.21(c), the LRA must include an evaluation of time-limited aging analyses (TLAAs). The applicant must demonstrate that (i) the analyses remain valid for the period of extended operation, (ii) the analyses have been projected to the end of the period of extended operation, or (iii) the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

Background

LRA Section 4.3.2 addresses the EAF TLAA for Class 1 piping systems. The 60-year projected CUF_{en} values for the limiting (bounding) EAF locations are described in LRA Table 4.3.1-2.

Issue

LRA Table 4.3.1-2 does not describe the materials of the limiting locations. The staff also needs clarification on whether the applicant eliminated certain EAF locations based on the more limiting EAF locations fabricated with a different material in the screening evaluation to determine the limiting EAF locations (e.g., a low alloy steel location was eliminated in consideration of the more limiting stainless steel location in the screening evaluation for EAF).

In addition, LRA Table 4.3.1-2 does not describe the specific piping systems or components (e.g., reactor vessel) of the limiting EAF locations.

Request

1. Provide the materials of fabrication for the limiting EAF locations listed in LRA Table 4.3.1-2.
2. Clarify whether the applicant eliminated certain EAF locations based on the more limiting EAF locations fabricated with a different material in the screening evaluation to determine the limiting EAF locations. If so, describe the eliminated EAF locations and discuss how the applicant determined the more limiting nature of the EAF locations fabricated with a different material (e.g., comparisons of F_{en} and CUF_{en} values).
3. Describe the specific piping systems or components (e.g., reactor vessel) of the limiting EAF locations in LRA Table 4.3.1-2.

Constellation Response:

1. The table in response to Request 3 below identifies the material of fabrication for all the EAF locations listed in LRA Table 4.3.1-2.

2. EAF locations were not eliminated based on a more limiting EAF location using a different material type. Locations were eliminated from further consideration per the methodology described in LRA Section 4.3.2.2 as summarized below.

   For each material type within a thermal zone, the location with the highest bounding screening CUFen value ($U_{en60scr}$) was selected and the location with the second highest $U_{en60scr}$ was also selected if the second component $U_{en60scr}$ value was greater than 1.0 and within a factor of 25 percent of the first component. In addition, components which are currently monitored by FatiguePro™ also screened in, regardless of the component's $U_{en60scr}$ value.

As a result, each "screened in" location bounds (for fatigue) "screened out" locations with the same material within the associated thermal zone. All "screened out" components within a thermal zone are represented by a least one "screened in" component of the same material type.

Therefore, this methodology does not allow a "screened in" component with one material type to be representative and bounding for a "screened out" component of a different material type.

3. The table below documents the applicable license renewal system and component type for all the EAF locations in LRA Table 4.3.1-2. Locations in LRA Table 4.3.1-2 which are not EAF locations are not shown in the table below. Note, reactor pressure vessel nozzles are designated as part of the Reactor Vessel System and not the associated process system.

EAF Limiting Locations From LRA Table 4.3.1-2				
No.	**Limiting EAF in LRA Table 4.3.1-2.**	**Material**	**License Renewal System**	**Component**
2	10" Nozzle-Shell Junction Element 169 (RPV_RRINNOZ)	LAS	Reactor Vessel	Recirc Inlet Nozzle
3	Vibration Instrument nozzle-Shell Junction (RPV_VIBNOZ)	LAS	Reactor Vessel	Recirc Outlet Nozzle
4	Miscellaneous Bracket Element 340 (RPV_MISCBRKT)	SS	Reactor Vessel	Bracket
5	CRD HSR Nozzle-Vessel Junction Element 217 (RPV_CRDHSRNOZ)	LAS	Reactor Vessel	CRD HSR Nozzle
6	Vessel at CRD Penetration, E 504 (RPV_ATCRDPEN)	LAS	Reactor Vessel	CRD Nozzle
7	FW Nozzle Safe End Element 228 (RPV_FWNSE-CS)	CS	Reactor Vessel	Feedwater Nozzle
8	Core Spray Nozzle Safe End Point 982 (RPV_CSNSE)	NBA	Reactor Vessel	Core Spray Nozzle
9	Core Spray Nozzle Safe End Ext. Point 61) (RPV_CSNSE-E)	CS	Reactor Vessel	Core Spray Nozzle
10	Liquid Control/DP Nozzle Safe End Element 456 (RPV_LCDPNSE)	SS	Reactor Vessel	Liquid Control/DP Nozzle
11	RHR/LPCI Nozzle Safe End Element 221 (RPV_LPCINSE)	NBA	Reactor Vessel	RHR/LPIC Nozzle

EAF Limiting Locations From LRA Table 4.3.1-2				
No.	**Limiting EAF in LRA Table 4.3.1-2.**	**Material**	**License Renewal System**	**Component**
12	RHR/LPCI Nozzle Element 42 (RPV_LPCINOZ)	CS	Reactor Vessel	RHR/LPIC Nozzle
13	1FW-01/02 Node 140B (FW_140B)	CS	Feedwater System	Piping
16	1MS-05 Node 345 (MS_345)	CS	Main Steam System	Piping
19	1MS-38A Node 85 at Valve 1B21-F067C (MS_85)	SS	Main Steam System	Piping
20	1RI-11 Node 160 Vent Side (RI_160)	CS	Reactor Core Isolation System	Piping
21	Node 735 RHR Tee to Valve (RR_735)	SS	Residual Heat Removal System	Piping
22	Node 60 Recirc Suction Pipe (RR_60)	SS	Reactor Recirculation System	Piping
23	1RR-32 Node 15 (RR_15)	SS	Reactor Recirculation System	Piping
24	1RT-01 Section C/F Node B470 (RT_B470)	CS	Reactor Water Cleanup System	Piping
26	1RH-03 Node 5RPV (RH_5RPV)	CS	Residual Heat Removal System	Piping
27	RHR/LPCI Penetration 1MC-17 (PEN_RHR)	CS	Residual Heat Removal System	Piping
38	FWNOZ_BR_N4AB	LAS	Reactor Vessel	Feedwater Nozzle
39	FWNOZ_BR_N4CD	LAS	Reactor Vessel	Feedwater Nozzle
40	FWNOZ_SE_N4AB	SS	Reactor Vessel	Feedwater Nozzle
41	FWNOZ_SE_N4CD	SS	Reactor Vessel	Feedwater Nozzle

Notes:
LAS – Low Allow Steel
CS – Carbon Steel
SS – Stainless Steel
NBA – Nickel Based Alloy

RAI 4.3.5-1

Regulatory Basis

Pursuant to 10 CFR 54.21(c), the LRA must include an evaluation of time-limited aging analyses (TLAAs). The applicant must demonstrate that (i) the analyses remain valid for the period of extended operation, (ii) the analyses have been projected to the end of the period of extended operation, or (iii) the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

Background

LRA Section 4.3.5 addresses the allowable stress and related high-energy line break (HELB) TLAAs for the piping systems designed in accordance with ASME Code Section III, Class 2, Class 3, and ANSI B31.1 design rules. LRA Table 4.3.5-2 describes the number of 60-year projected cycles for each non-Class 1 piping system to confirm that the 60-year projected cycles do not exceed 7000 cycles in the implicit fatigue analysis.

Issue

However, LRA Section 4.3.5 does not clearly describe how the 60-year cycles were determined (e.g., based on piping system design information, plant operation procedures, test requirements, USAR information and specific system-level knowledge).

Request

Clarify how the applicant estimated the 60-year cycles for the non-Class 1 piping systems (e.g., based on piping system design information, plant operation procedures, test requirements, USAR information and specific system-level knowledge).

Constellation Response:

Non-Class 1 piping potentially subject to thermal cycling was identified from the site configuration database, i.e., Passport. Piping segments with design temperatures of 220 degrees F or greater were identified. The piping segments are identified by the two-letter system designator codes used by Clinton, as identified on piping and instrumentation drawings (P&IDs) and license renewal basis document CL-SSBD-SSL, “License Renewal Systems and Structures Scoping and Screening Basis Document.” Individual system scoping documents and LR boundary drawings were then used to determine which piping segments were associated with implicit fatigue analyses per the following criteria:

1) The piping system/segment is in scope of license renewal,
2) The piping system/segment was designed as ASME Section III, Class 2, Class 3, or ANSI B31.1, and
3) The piping system/segment has an operating temperature greater than 220 degrees F for carbon steel and 270 degrees F for stainless steel.

Operating temperatures were based on Passport data for individual piping and valves in the piping segment. In addition, USAR Table 9.3-3 was also used as a source to determine which in-scope sample piping segments have operating temperatures greater than 220 degrees F.

In scope piping segments that met the above criteria were then investigated to estimate conservative cycle projections for 60-years per the following.

1) An assessment if the piping segment heats up and cools down with the RPV, in which case all the projections in LRA 4.3.1-1 were assumed.
2) For systems that do not heat up and cool down with the RPV, projections were based on:
    a. System procedures,
    b. Surveillance testing and conservative estimates based on required Technical Specification frequencies,
    c. USAR operational requirements for the system,
    d. Discussions with operations and chemistry,
    e. Passport information related to testing,
    f. Discussions with system engineers.

RAI 4.3.1-1

Regulatory Basis

Pursuant to 10 CFR 54.21(c), the LRA must include an evaluation of time-limited aging analyses (TLAAs). The applicant must demonstrate that (i) the analyses remain valid for the period of extended operation, (ii) the analyses have been projected to the end of the period of extended operation, or (iii) the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

Background

LRA Table 4.3.1-1 indicates that the number of the 60-year projected occurrences (cycles) of the "design hydrostatic test" transient is 44, which is greater than the design transient cycles of 40. Similarly, LRA Table 4.3.1-1 indicates that, for the "turbine roll" transient and "HOTZERO – Hot Zero Power Scram" transient, the number of 60-year projected cycles exceed the number of design transient cycles.

Issue

In comparison, the applicant dispositioned the fatigue TLAAs for the following components in accordance with 10 CFR 54.21(c)(1)(i): (1) main steam isolation valves; (2) safety/relief valves; (3) recirculation system flow control valves; (4) recirculation system gate valves; (5) recirculation system pumps; (6) control rod drives; and (7) core plate stiffener to skirt weld and top guide/grid reactor vessel internal components.

Given the TLAA disposition for the components discussed above in accordance with 10 CFR 54.21(c)(1)(i) (i.e., not using cycle projections or the Fatigue Monitoring AMP), the staff needs clarification on whether the 60-year projected cycles of the "design hydrostatic test" transient, "turbine roll" transient and "HOTZERO – hot zero power scram" transient, which are greater than the design cycles, may affect the validity of the fatigue TLAA disposition (e.g., resulting in the 60-year CUF values of these components exceeding the design limit).

Request

Given the TLAA disposition for the components discussed in the issue section in accordance with 10 CFR 54.21(c)(1)(i) (i.e., not using cycle projections or the Fatigue Monitoring AMP), clarify whether the 60-year projected cycles of the "design hydrostatic test" transient, "turbine roll" transient and "HOTZERO – hot zero power scram" transient, which are greater than the design cycles, may affect the validity of the fatigue TLAA disposition (e.g., resulting in the 60-year CUF values of these components exceeding the design limit).

Constellation Response:

The transient projections in LRA Table 4.3.1-1 were compared to the transient occurrences assumed in the fatigue TLAAs described in LRA Sections 4.3.3.2, 4.3.3.3, 4.3.3.5, 4.3.3.6, 4.3.3.7, 4.3.3.8, and 4.3.7.1 to support their 10 CFR 54.21(c)(1)(i) disposition. These are the LRA sections that describe, evaluate, and disposition the components listed in the above issue discussion.

The 60 year projected transient occurrences of only three transients in Table 4.3.1-1 exceed the number of design occurrences. These are:

- The “Design Hydrostatic Test” transient with 44 projected occurrences over 60 years versus 40 design occurrences.
- The “Turbine Roll” transient with 123 projected occurrences over 60 years versus 120 design occurrences. And
- The “HOTZERO – Hot Zero Power Scram” transient with 21 projected occurrences over 60 years versus 17 design occurrences.

The table below documents if the above three transients were assumed in the fatigue evaluations described in LRA Sections 4.3.3.2, 4.3.3.3, 4.3.3.5, 4.3.3.6, 4.3.3.7, 4.3.3.8, and 4.3.7.1. The table also documents the number of assumed occurrences if the transient was assumed in the fatigue evaluation.

As is demonstrated in the table, the “Turbine Roll” and “HOTZERO – Hot Zero Power Scram” transients were not considered or assumed in any of the fatigue evaluations described in LRA Sections 4.3.3.2, 4.3.3.3, 4.3.3.5, 4.3.3.6, 4.3.3.7, 4.3.3.8, and 4.3.7.1. Therefore, the impact of the exceedance of these two projected transients has no impact on the corresponding fatigue evaluations described, evaluated, and dispositioned in LRA Sections 4.3.3.2, 4.3.3.3, 4.3.3.5, 4.3.3.6, 4.3.3.7, 4.3.3.8, and 4.3.7.1; and the 10 CFR 54.21(c)(1)(i) disposition is appropriate.

In the case of the “Design Hydrostatic Test” transient, the table shows that this transient was assumed in only the fatigue evaluations of the safety/relief valves (LRA Section 4.3.3.3), the recirculation system flow control valves (LRA Section 4.3.3.5), recirculation system gate valves (LRA Section 4.3.3.6), and control rod drive system (LRA Section 4.3.3.8). The assumed occurrence values in these fatigue evaluations were either 120 or 130 occurrences. Comparison of these assumed values versus the 60 year projected value of 44, supports the 10 CFR 54.21(c)(1)(i) disposition.

LRA Section 4.3 Components with 10 CFR 54.21(c)(1)(i) Dispositions	**Design Hydrostatic Test**	**Turbine Roll Transient**	**HOTZERO Power Scram Transient**
4.3.3.2 Main Steam Isolation Valves	Transient was not assumed in the fatigue evaluation.	Transient was not assumed in the fatigue evaluation.	Transient was not assumed in the fatigue evaluation.
4.3.3.3 Safety/Relief Valves	120 Hydrostatic Tests were assumed versus 44 are projected.	Transient was not assumed in the fatigue evaluation.	Transient was not assumed in the fatigue evaluation.
4.3.3.5 Recirculation System Flow Control Valves	130 Hydrostatic Tests were assumed versus 44 are projected.	Transient was not assumed in the fatigue evaluation.	Transient was not assumed in the fatigue evaluation.

LRA Section 4.3 Components with 10 CFR 54.21(c)(1)(i) Dispositions	**Design Hydrostatic Test**	**Turbine Roll Transient**	**HOTZERO Power Scram Transient**
4.3.3.6 Recirculation System Gate Valves	130 Hydrostatic Tests were assumed versus 44 are projected.	Transient was not assumed in the fatigue evaluation.	Transient was not assumed in the fatigue evaluation.
4.3.3.7 Recirculation System Pumps	Transient was not assumed in the fatigue evaluation.	Transient was not assumed in the fatigue evaluation.	Transient was not assumed in the fatigue evaluation.
4.3.3.8 - Control Rod Drive System	130 Hydrostatic Tests were assumed versus 44 are projected.	Transient was not assumed in the fatigue evaluation.	Transient was not assumed in the fatigue evaluation.
4.3.7.1 Core Plate Stiffener to Skirt Weld and Top Guide/Grid RVI components	Transient was not assumed in the fatigue evaluations.	Transient was not assumed in the fatigue evaluation.	Transient was not assumed in the fatigue evaluation.

RAI 4.7.5-1

Regulatory Basis

Pursuant to 10 CFR 54.21(c), the LRA must include an evaluation of time-limited aging analyses (TLAAs). The applicant must demonstrate that (i) the analyses remain valid for the period of extended operation, (ii) the analyses have been projected to the end of the period of extended operation, or (iii) the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

Background

LRA Section 4.7.5 addresses the fatigue TLAA for hydraulic control units (HCUs). The LRA also explains that the USAR Section 3.9.1.1.3, "Hydraulic Control Unit Transients," documents the transients and transient occurrences that were considered in the design of the HCUs.

Issue

The staff noted that USAR Section 3.9.1.1.3 indicates that the following design cycles related to scrams were evaluated in the existing fatigue analysis for the HCUs: (1) 140 cycles of the "scram test" transient; (2) 160 cycles of "startup scram" transient; and (3) 300 cycles of "operational scram" transient. However, LRA Section 4.7.5 does not clearly describe the 60-year projected cycles for these scram transients compared to the transient cycles evaluated in the existing fatigue analysis.

In addition, LRA Section 4.7.5 indicates that USAR Section 3.9.2.2.1.6.4 "Hydraulic Control Unit (HCU)," documents that the HCUs were analyzed for faulted conditions including the effects of seismic and hydrodynamic loads. The LRA explains that this design adequacy was determined by testing and analysis and that the qualification testing included vibration testing equivalent to 1800 safety relief valve (SRV) actuations, one operational basis earthquake (OBE), and one safety shutdown earthquake (SSE). However, LRA Section 4.7.5 does not clearly describe the 60-year projected cycles (occurrences) of the SRV actuations.

Request

1. Describe the 60-year projected cycles of the following scram-related transients to confirm that the 60-year projected cycles are bounded by (less than) the cycles evaluated in the existing fatigue analysis for the HCUs: (1) "scram test" transient; (2) "startup scram" transient; and (3) "operational scram" transient. As part of the discussion, explain the basis of these cycle projections.
2. Describe the 60-year projected cycles of the SRV actuations to confirm that the projected cycles are bounded by (less than) the SRV actuation cycles (i.e., 1800 cycles) evaluated in the fatigue analysis for the effect of seismic and hydrodynamic loads on the hydraulic control units. As part of the discussion, explain the basis of the cycle projection.

Constellation Response:

1. Each of the transients listed in Request 1 is described and evaluated separately below.

“Scram Test”
The contribution to fatigue from the “Scram Test” transient is insignificant and, therefore, 60 year projections of this transient is not necessary. The basis for this conclusion is discussed below.

The control rod drive system design specification documents the transients that the design of the system should consider. The HCUs are part of the control rod drive system. The specified transients and transient occurrences are the same as those documented in USAR Section 3.9.1.1.3, with the exception that the design specification requires the consideration of 10 “Vessel Overpressure” transient occurrences. This transient is an “emergency” condition and is, therefore, not addressed in LRA Section 4.7.5.

The design specification documents that the “Scram Test” transient requirement applies to unpressurized vessel conditions. Under these conditions reactor coolant temperature returning to the HCUs from the vessel would be less than 212 degrees F and any temperature changes during this transient would be very small and less than 212 degrees F and result in no contribution to fatigue.

“Startup Scram”
The design specification documents that the “Startup Scram” transient requirements apply to vessel temperatures between 100 degrees F and 400 degrees F. Under these conditions, temperatures returning to the HCUs from the vessel would also be in this range and any temperature changes during this transient would be small and the result on fatigue is concluded to be insignificant.

As documented in LRA Section 4.7.5, except for the accumulators and nitrogen tanks, pressure retaining components on the HCUs were designed per ANSI B31.1.0 criteria. As such, there are no explicit fatigue calculations for the HCUs. The ANSI B31.1.0 criteria of 7,000 full temperature occurrences assumes that each occurrence will heatup and pressurize the component from ambient conditions to the design temperature and pressure. The design temperature and pressure of the control rod drive system is 575 degrees F and 1250 psi. Therefore, the severity of the temperature transients during the “Startup Scram” transient is much less than the severity assumed in the ANSI B31.1.0 criteria.

The design specification required the evaluation of the following normal transients for the control rod drive system: Startup/Shutdown (from LRA Table 4.3.1-1 with the 60 year projection of 101 occurrences), Operational Scrams (from LRA Table 4.3.1-1 items 10, 11, 20, and 21, with the 60 year projection of less than 90 occurrences), OBE (from LRA Table 4.3.1-1 with the 60 year projection of 1 with 10 peaks). The remaining transients specified in the design specification, including the “Scram Test” transient are concluded to have no contribution to fatigue. Therefore, with the exception of the Startup Scram transient, the total number of transients which contribute to fatigue that were specified in the design specification are projected to occur less than 201 times in 60 years.

The remaining 6,799 cycles, in which components are heated from ambient conditions to the design temperature (more than a 500 degrees F temperature difference), is margin which bounds the insignificant contribution of the “Startup Scram” transient in which the temperature changes are less than 300 degrees F. Conservative estimates of how many times each HCU may experience the “Scram Test” transient during plant startup and during normal operations is concluded to be significantly less than 50 times per year, or 3,000 occurrences over 60 years.

Operational Scram
The “Operational Scram” transient may contribute to fatigue and was, therefore, projected for 60 years. “Operational Scram” projections are documented in LRA Table 4.3.1-1 as transients 10, 11, 20, and 21. LRA Table 4.3.1-1 shows that the total number of occurrences projected for 60 years for these four transients is less than 90 occurrences.

2. The design adequacy testing described in LRA Section 4.7.5 included vibration testing for seismic and hydrodynamic loads equivalent to 1,800 SRV actuations, one OBE, and one SSE. LRA Table 4.3.1-1 documents that all SRVs are projected to actuate 300 times in 60 years. LRA Table 4.3.1-1 also documents that Clinton is projected to experience one OBE and one SSE in 60 years. Therefore, comparison of the projected 300 occurrences over 60 years for all SRVs to the 1,800 SRV actuations simulated by the vibration testing shows that the testing remains valid for 60 years.

RAI 4.3.7-1

Regulatory Basis

Pursuant to 10 CFR 54.21(c), the LRA must include an evaluation of time-limited aging analyses (TLAAs). The applicant must demonstrate that (i) the analyses remain valid for the period of extended operation, (ii) the analyses have been projected to the end of the period of extended operation, or (iii) the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

Background

The LRA explains that the 40-year CUF for the core shroud support plate, which is the limiting location for the core shroud support structure and core shroud stabilizer assembly, is 0.426. The LRA indicates that the CUF value is based on safety relief valve (SRV) actuation transient cycles, which are greater than 12000 cycles in the 40-yearfatigue analysis. The LRA also explains that the SRV actuation transient is the most significant contributor to the fatigue in the core shroud support plate.

Issue

In contrast, the following General Electric report indicates that the most significant contribution to the 40-year CUF of the core shroud support plate is due to thermal cycles (i.e., contribution of 0.406 due to certain thermal cycles) (Reference: GE-NE-26A-6217, "Shroud and Shroud Support Structure," Section 6.2.5.2, March 9, 2005). The reference also indicates that the calculation of the CUF contribution (0.406) is based on the maximum usage factor in the shroud support plate for a similar standard BWR/6 plant.

The reference further explains that the 40-year CUF contribution of the SRV actuation cycles to the core shroud support plate is approximately 0.013 and that the 40-year CUF contribution of other thermal transients is less than 0.01.

However, the LRA does not clearly discuss the following items related to the maximum CUF contribution (0.406) due to certain thermal cycles: (1) specific thermal transients and their cycles evaluated in the CUF calculation for a standard BWR/6 plant; (2) whether the transient cycles evaluated for the standard plant reasonably represent the 40-year transient cycles of the Clinton Power Station.

Request

1. Given the maximum CUF contribution due to thermal cycles (non-SRV-actuation cycles) for the core shroud support plate discussed in the reference in the issue section, clarify whether the SRV actuation transient is the most significant contributor to the 40-year CUF of the core shroud support plate. If not, describe the following information: (1) the transients and their 40-year cycles that make the most significant contribution to the CUF and (2) the most significant contribution of these transients to the CUF (i.e., the partial CUF due to these transient cycles).
2. Clarify whether the transient cycles, which make the most significant CUF contribution to the core shroud support plate, reasonably represent the 40-year cycles of the Clinton Power Station in terms of the fatigue analysis for the core shroud support plate.

3. Revise the LRA as needed based on the discussion above.

Constellation Response:

1. The 12,600 SRV cycles assumed in the 40-year fatigue evaluation described and evaluated in LRA Section 4.3.7.3 is the most significant contributor to the calculated 40 year design CUF value.

   Prior to 2006, the design CUF for the core shroud support plate was a value of 0.406 over 40 years. This CUF total value was a combination of [[ ]], for a CUF subtotal of [[ ]], plus New Load Design Adequacy (NLDA) with a CUF subtotal of [[ ]], for a total of 0.406. The NLDA CUF adder considered SRV loads and assumed [[ ]].

   In 2006, Clinton installed core shroud stabilizer assembly brackets. The modification altered the load path and produced additional loads on the core shroud support plate and the previous CUF value of 0.406 was revised to 0.426 in the General Electric proprietary report referenced above (GE-NE-26A-6217, "Shroud and Shroud Support Structure"). This increase included a [[ ]].

   The [[ ]] are reflected in the pre-modification NLDA adder of [[ ]], and the [[ ]] added by the analysis for the modification, bringing the CUF subtotal value for [[ ]] to [[ ]], which is [[ ]] of the total CUF value of 0.426. Therefore, the assumption of [[ ]] is the most significant contributor to the 40-year CUF value of the core shroud support plate.

2. The response to Request 1 above documents the basis for why [[ ]] assumed over 40 years is the most significant contributor to the core shroud support plate fatigue. LRA Section 4.3.7.3 did not address the remaining thermal transients that were assumed in the fatigue evaluation because the contribution to fatigue for each of these transients projected for 60 years is less than assumed in the analysis.

   To clarify, the table below documents the transients that make up all the thermal load pairs used in the original CUF calculation of [[ ]] over 40 years. The second column in the table documents the number of occurrences assumed for 40 years and the fourth column documents the projected number of occurrences for 60 years from LRA Table 4.3.1-1.

Transients Assumed in the Original Fatigue Evaluation.	Number of Occurrences Assumed Over 40 Years in the Fatigue Evaluation	Transient Numbers on LRA Table 4.3.1-1.	Number of Projected Occurrences Over 60 Years, From LRA Table 4.3.1-1
[[		2	44
		3	101
		10, 11, 22	82
		12, 41	1
		15,17	98
		20	5
		21	3
		23	1
		24	1
	]]	26	1

In all cases the projected number of occurrences for 60 years is less than or equal to the number of occurrences assumed for 40 years in the fatigue evaluation, which indicates that the projected 60-year CUF subtotal for the transients above will be less than a CUF value of [[ ]]. If the actual number of occurrences for any of the above transients exceeds the originally assumed number of occurrences, the impact on the actual CUF value will be small. Therefore, it is not credible that these transients would result in the actual CUF value exceeding the ASME limit of 1.0 during the period of extended operation.

Additionally, the core shroud support plate is monitored by FatiguePro™ as documented in item number one in LRA Table 4.3.1-2, which shows a projected CUF value for 60 years of 0.265. The FatiguePro™ software counts all the above transients and [[ ]] to calculate a CUF value utilizing the methodology in the fatigue evaluations.

3. No updates to the LRA are required as a result of this response.

RAI 4.6.1-1

Regulatory Basis

10 CFR 54.21(a)(3) requires an applicant to demonstrate that the effects of aging for structures and components will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation. As described in the SRP-LR, an applicant may demonstrate compliance with 10 CFR 54.21(a)(3) by referencing the GALL-LR Report when evaluation of the matter in the GALL-LR Report applies to the plant.

Background

LRA Section 4.6.1 provides TLAA evaluations for two types of transients: monitored transients and unmonitored transients, claiming consistency with 10 CFR 54.21(c)(1)(i). However, it is not clear whether this is consistent with SRP-LR Section 4.6.3.1.1.1. Per SRP-LR Section 4.6.3.1.1.1, the number of assumed transients in the existing analysis needs to be compared with the extrapolation to 60 years of operation of the number of operating transients experienced to date. It is not clear how monitored and unmonitored transients are related to operating transients experienced to date and whether unmonitored transients have never happened to the plant.

Request

1. Clarify how unmonitored transients are related to transients the plant has experienced to date.
2. Confirm that the calculations tallied in the fourth column of LRA Table 4.6.1-1 considered both monitored and unmonitored transients that were applicable to each penetration calculation. Or confirm that the monitored transients listed in LRA Table 4.6.1-1 are only applicable transients in calculations considered in the fourth column of LRA Table 4.6.1-1 for the sake of cumulative fatigue analysis.
3. The third column of LRA Table 4.6.1-2 provides the type and number of assumed unmonitored transients applicable to each group of penetrations listed in the first column of LRA Table 4.6.1-2. Confirm that there are no other monitored transients that may be applicable to each group of penetrations.

Constellation Response:

1. The unmonitored transients documented in LRA Section 4.6.1 were originally specified during the initial unit design phase based on postulated specific transients for the process piping contained in the associated penetration. Therefore, the original penetration fatigue calculations assumed transients based on the associated system functions and testing requirements. The number of transient occurrences were conservatively assumed based on how the systems were expected to operate and be tested at the time of design. Because of uncertainty, the assumed number of occurrences were purposefully selected to be very conservative. For example: some of the transient occurrences were based on the assumption that Clinton would refuel every year for 40 years, assumed transient occurrences were based on 275 main fire pump starts per year over 40 years, or system overpressure relief valves would actuate once per year for 40 years.

These transients are controlled and observed by operations in accordance with site procedures. However, the actual number of occurrences of most of these transients have not been recorded.

For license renewal, Constellation reviewed the number of occurrences assumed for each transient in each calculation to ascertain if the assumptions were still conservative and valid for 60 years. As explained in LRA Section 4.6.1, the review found only two out of 121 calculations that required revision for 60 years of operations.

For the 60 year projections in LRA Table 4.6.1-2, Constellation used the same methodology to conservatively project the number of occurrences over 60 years. However, Constellation took advantage of a better understanding of how the systems are operated and tested. In addition, equipment history records were consulted to understand past component failures. For example: Clinton refuels every two years, main fire pump starts are less than 83 times per year and based on equipment history the associated overpressure relief valves have not actuated since Clinton started up. This has allowed more precise but conservative projections of the number of occurrences over 60 years.

The basis for the 60 year projections in LRA Table 4.6.1-2 are described below. Note, as requested in the RAI, these sections also document estimates of transient occurrences that Clinton has experienced as of June 30, 2022, based on the same methodology.

Penetrations 1MC-11, 1MC-12, 1MC-13, 1MC-18, 1MC-19, 1MC-20
These penetrations are associated with the three residual heat removal (RHR) pump suction and test lines to the suppression pool. The original fatigue evaluations assumed 480 “Pump Operation” and “Pump Cooldown” transient occurrences in which each penetration heats up from 70 degrees F to 120 degrees F in 30 seconds, remains at 120 degrees F for 30 minutes, and then cools down from 120 degrees F to 70 degrees F in 30 hours. This temperature profile in the original calculations was based on the following operational events: 1) the associated RHR pump is tested, 2) the associated RHR pump is placed into minimum flow bypass, and 3) the associated RHR loop is placed in suppression pool cooling mode.

Review of RHR system operational modes and operating experience reveals that these penetrations will most likely never experience the above specified temperature transient. This is because:

1) These penetrations contain process piping that transfer suppression pool water, using the associated RHR pump, in and out of the suppression pool.
2) Technical Specification 3.6.2.1 requires suppression pool temperature to be less than 95 degrees F when reactor power is greater than one percent and no testing that adds heat to the suppression pool is being performed.
3) There is no heating source in the flow path which could elevate the temperature of the fluid in these penetrations to 120 degrees F.

Therefore, the RHR pump test and RHR pump minimum flow bypass operational events cannot result in the temperature transient described above and are, therefore, not counted toward the specified 480 “Pump Operation” and “Pump Cooldown” transient occurrences.

With respect to the suppression pool cooling operational event, Technical Specification 3.6.2.1 requires suppression pool temperature to be less than or equal to 105 degrees F when reactor power is greater than one percent and testing that adds heat to the suppression pool is being performed. Reactor core isolation cooling (RCIC) turbine driven pump testing may result in suppression pool temperatures in this range.

The RCIC turbine driven pump surveillance test does add a small amount of heat to the suppression pool. This is because the turbine exhausts to the suppression pool, which causes the temperature in the pool to stratify. To eliminate the stratification by mixing the water in the suppression pool, one RHR system loop is run in suppression pool cooling mode. During the initial few minutes when the RHR system loop is placed in service, temperatures in the associated penetrations may reach temperatures in the range of 95 degrees F to 105 degrees F.

As a result, this event is conservatively credited in this TLAA towards the “Pump Operation” and “Pump Cooldown” transient occurrences assumed in the original associated fatigue calculations. Typically, only one RHR loop is placed in service. However, for conservatism, the assumption is made that all three RHR loops are placed in service after the RCIC turbine driven pump is tested. Since the RCIC turbine driven pump is tested quarterly this results in a total of 240 occurrences over 60 years. An additional 100 occurrences over 60 years were added for margin, for a total of 340 occurrences in 60 years. As of June 30, 2022, Clinton has experienced approximately 136 RCIC turbine driven pump surveillances.

Penetrations 1MC-15 and 1MC-16
These penetrations contain RHR system piping for the low pressure coolant injection (LPCI) mode of the RHR system. Each LPCI loop is tested every two years per the IST Program and in addition LPCI is projected to actuate one time in 60 years for a total of 31 occurrences in 60 years. A projection of 100 occurrences over 60 years was chosen for conservatism. As of June 30, 2022, Clinton has experienced zero LPCI injections. Therefore, the estimated number of occurrences as of June 30, 2022, is 17 occurrences.

Penetrations 1MC-24, 1MC-26, 1MC-38, and 1MC-31
Note the “Projected Occurrences In 60 Years” for “Relief Valve Actuations” in LRA Table 4.6.1-2 for penetrations 1MC-24, 1MC-26, 1MC-38, and 1MC-31 were revised from “Significantly less than 40” to “5” in LRA Supplement 1, dated November 27, 2024, via Change #37.

These penetrations contain piping which direct flow from overpressure relief valve discharge piping to the suppression pool. These relief valves protect heat exchangers or pumps from system over pressure conditions. These events are infrequent. The associated calculations assumed that each valve would actuate due to an actual over pressure condition or a spurious actuation 40 times in 40 years. The actuation of these valves is an off normal condition which would be entered into the corrective action program and the underlying cause would be corrected. A review of the Clinton equipment history database for all the relief valves associated with these penetrations indicates, as of June 30, 2022, that there is no evidence that any of these valves have actuated since initial plant startup. Therefore, there is reasonable assurance that the expected number of actuations would not exceed five occurrences after June 30, 2022.

Penetrations 1MC-52 and 1MC-53
The process piping in penetrations 1MC-52 and 1MC-53 direct flow to and from the fuel pool cooling and cleanup system to the reactor vessel pool, reactor vessel steam dryer storage pool, and the reactor vessel steam separator storage pool during refueling outages. The associated calculations assumed 40 refueling outages associated with placing the piping in these penetrations in service during a refueling outage. Clinton's 40-year license expires in April 2027 before refueling outage number 23 in September 2027. Assuming there will be an additional 11 refueling outages in the period of extended operation results in a total of 34 refueling outages. The current number of occurrences as of June 30, 2022, is estimated as 21.

Penetrations 1MC-56, 1MC-81, and 1MC-82
These penetrations contain fire protection system piping which supply fire water from the main fire protection system yard header to fire protection system piping inside containment.

The original calculation assumed that the piping in these penetrations would experience a flow transient resulting from a relief valve actuation in containment each time one of the system main fire pumps is started. This is a very conservative assumption since most of the time when a system main fire pump starts, it supplies flow to other portions of the fire protection system outside containment, and a relief valve actuation does not occur. Directed fire suppression system flow to inside containment is a very rare event. However, because of this assumption the original calculations assumed 11,000 transient occurrences (or 275 every year) for each penetration over 40 years.

This methodology was also used for the 60 year projections for these penetrations in LRA Table 4.6.1-2. However, a detailed review of the required surveillances for the main fire pumps revealed that the assumption of 11,000 pump starts was extremely conservative.

Based on existing surveillance testing frequencies including pump and system functional and flow testing, the main fire pumps will start an estimated 3,560 times in 60 years. In addition, 624 starts were estimated during construction, and an additional 800 miscellaneous pump starts are assumed for additional conservatism. This results in a total estimate of 4,992 pump starts over 60 years. Based on these estimates the number of pumps starts as of June 30, 2022, are conservatively estimated as no more than 3,100 occurrences.

Penetrations 1MC-34 and 1MC-79
The process piping in these penetrations direct flow in the suppression pool cleanup system to and from the suppression pool. The associated calculations for these penetrations assumed the suppression pool cleanup system is placed in and out of service 2,080 times over 40 years. This is extremely conservative. The system procedure documents that this system is placed in service to reduce the radioactivity level of the suppression pool. This system is typically placed in service only during refueling outages. Therefore, a more appropriate but conservative estimate is that this system would be placed in service no more than 10 times a year or 600 times in 60 years. The current number of occurrences as of June 30, 2022, is estimated as 340.

Penetration 1MC-69
The process piping in this penetration directs flow from the containment floor drain sumps out of containment. The original calculation assumed an average containment identified leak

rate of 5.68 gpm continuously for 40 years. Based on a continuous 5.68 gpm rate, the size of the sump, and the pump flow capacity, this assumption results in 45 pump starts per day or a total of 657,000 cycles in 40 years. This is an extremely conservative assumption since Clinton operating experience indicates an identified leak rate of less than 3.5 gpm.

Assuming Clinton operates continuously for 60 years with an identified leak rate of 3.5 gpm results in approximately 607,129 cycles. The number of occurrences as of June 30, 2022, is estimated to be less than 344,040.

Penetration 1MC-70
The process piping in this penetration directs flow from the drywell floor drain sumps out of containment. The original calculation assumed an average containment unidentified leak rate of 2.68 gpm continuously for 40 years. Based on a continuous 2.68 gpm rate, the sump size, and the pump flow capacity this assumption would result in 24 pump starts per day or a total of 350,400 cycles in 40 years. This is a conservative assumption since Clinton operating experience indicates an unidentified leak rate of less than 1.0 gpm.

Assuming Clinton operates continuously for 60 years with an unidentified leak rate of 1.0 gpm results in approximately 197,100 cycles. The current number of occurrences as of June 30, 2022, is estimated to be less than 111,690.

Penetration 1MC-32
The process piping in this penetration directs flow from the suppression pool to the low pressure core spray (LPCS) system. The original calculation assumed 480 occurrences over 40 years based on assumed system surveillances.

The LPCS system pump is tested through penetration 1MC-32 quarterly per the IST Program and in addition the LPCS system is projected to actuate one time in 60 years for a total of 241 occurrences over 60 years. The 60 year projection was rounded up to 300 occurrences in the 60-year life of the plant. The current number of occurrences as of June 30, 2022, is estimated as 137.

Penetration 1MC-33
40 “Relief Valve Blow Downs”
Note the “Projected Occurrences In 60 Years” for “Relief Valve Actuations” in LRA Table 4.6.1-2 for penetration 1MC-33 was revised from “Significantly less than 40” to “5” in LRA Supplement 1, dated November 27, 2024, via Change #37.

This penetration contains piping which directs flow from overpressure relief valve discharge piping to the suppression pool. These relief valves protect tanks and pumps from system overpressure conditions. These events are infrequent. The associated calculations assumed that the valves would actuate due to an actual overpressure condition or a spurious actuation 40 times in 40 years. The actuation of these valves is an off normal condition which would be entered into the corrective action program and the underlying cause would be corrected. A review of the Clinton equipment history database for these relief valves indicates, as of June 30, 2022, that there is no evidence that any of these valves have actuated since initial plant startup. Therefore, there is reasonable assurance that the expected number of actuations would not exceed five occurrences after June 30, 2022.

40 "HPCS System Test" Transients
The process piping in penetration 1MC-33 directs flow from the high pressure core spray (HPCS) system to containment when the pumps are tested during refueling outages. The original calculation assumed 40 HPCS system tests in 40 years. Plant procedures prohibit the system to be tested through penetration 1MC-33 when the plant is at power.

Therefore, 30 tests are projected over 60 years, and LRA Table 4.3.1-1 documents that this system is projected to actuate seven times in 60 years for a total of 37. The current number of occurrences as of June 30, 2022, is estimated as 21.

Penetration 1MC-74
Penetration 1MC-74 is attached to blind flanges on both sides of the penetration. Therefore, there is no actual process piping attached to either side. However, this penetration may be used during decontamination activities during a refueling outage. The original calculation assumed 20 occurrences in 40 refueling outages over 40 years for a total of 800 occurrences.

Clinton is projected to experience 34 refueling outages over 60 years. Conservatively assuming 20 occurrences per refueling outage results in a 60 year projection of 680 occurrences. The current number of occurrences as of June 30, 2022, is conservatively estimated to be 385.

Penetrations 1MC-64 and MC-65
Review of the original calculations for these penetrations showed that some assumed unmonitored transient occurrences over 40 years were not valid for 60 years. These issues were entered into the Clinton corrective action program and corrected as follows.

For the calculation associated with penetration 1MC-64, the "RT Heat Exchanger Swap" transient which was assumed to occur 20 times in 40 years was revised to 600 occurrences over 60 years. The resulting CUF value for 60 years was 0.272.

For the calculation associated with penetration 1MC-65, the "Backwash" transient which was assumed to occur 2,450 times in 40 years was revised to 110,000 occurrences over 60 years. The resulting CUF value for 60 years was 0.110.

2. Ninety-five of the 121 calculations assumed only the monitored transients shown in LRA Table 4.3.1-1 or transients that specified no temperature or pressure changes (i.e., steady state conditions). Each monitored transient is documented in the first column of LRA Table 4.6.1-1. These same 95 calculations did not assume any of the "unmonitored" transients documented in LRA Table 4.6.1-2.

   The fourth column of LRA Table 4.6.1-1 represents the number of calculations of the 95 that assumed the transient in the first column. For example, the first row of LRA Table 4.6.1-1 documents that 87 of the 95 calculations assumed 123 Boltup transient occurrences. This does not mean that the 87 calculations only assumed 123 Boltup transient occurrences. Some of these 87 calculations may have also assumed other monitored transients in LRA Table 4.6.1-1. For example, the calculation for penetration 1MC-14 assumed the following transients which are in LRA Table 4.6.1-1: Boltup, Hydro Test, Leak Test, Startup, Turbine Trip, Composite Loss of Feedpump, Unbolt, and Refuel.

3. The calculations for six penetrations in LRA Table 4.6.1-2 assumed monitored transients documented in Table 4.6.1-1. The impact of these assumed monitored transients have an insignificant or no impact on penetration fatigue calculations.

   The calculations associated with the penetrations in LRA Table 4.6.1-2 assumed:

   1) The unmonitored transients shown in LRA Table 4.6.1-2,
   2) Transients that specified insignificant temperature changes (e.g., less than 100 degrees F delta). These are identified and described below, and
   3) Several monitored transients described in LRA Table 4.6.1-1 which contribute insignificantly to fatigue. These are identified and described below.

   A complete breakdown of the transients not shown in LRA Table 4.6.1-2, reflected in items 2 and 3 above, and the impact on the applicable penetrations is provided below.

   Transients That Specified Insignificant Temperature Changes
   The associated calculations for 10 of the penetrations in LRA Table 4.6.1-2 considered transients that specified insignificant temperature changes (e.g., less than 100 degrees F) in which the resulting impact on fatigue is insignificant. Because of the insignificant impact of these transients on fatigue for the component, the calculations did not actually calculate a subtotal CUF value based on the transients. These transients were not documented in LRA Table 4.6.1-2. The following detailed explanation illustrates this subset.

   The calculation associated with penetration 1MC-15 documents that the only specified transients for this calculation are: 1) 10,000 Shut-Down Cooling transients where process temperature changes from 70 degrees F to 65 degrees F, and 2) 222 “Pump Operation” transients where process temperature changes from 65 degrees F to 100 degrees F and back to 65 degrees F. An excerpt of the transient specification from the calculation is shown below.

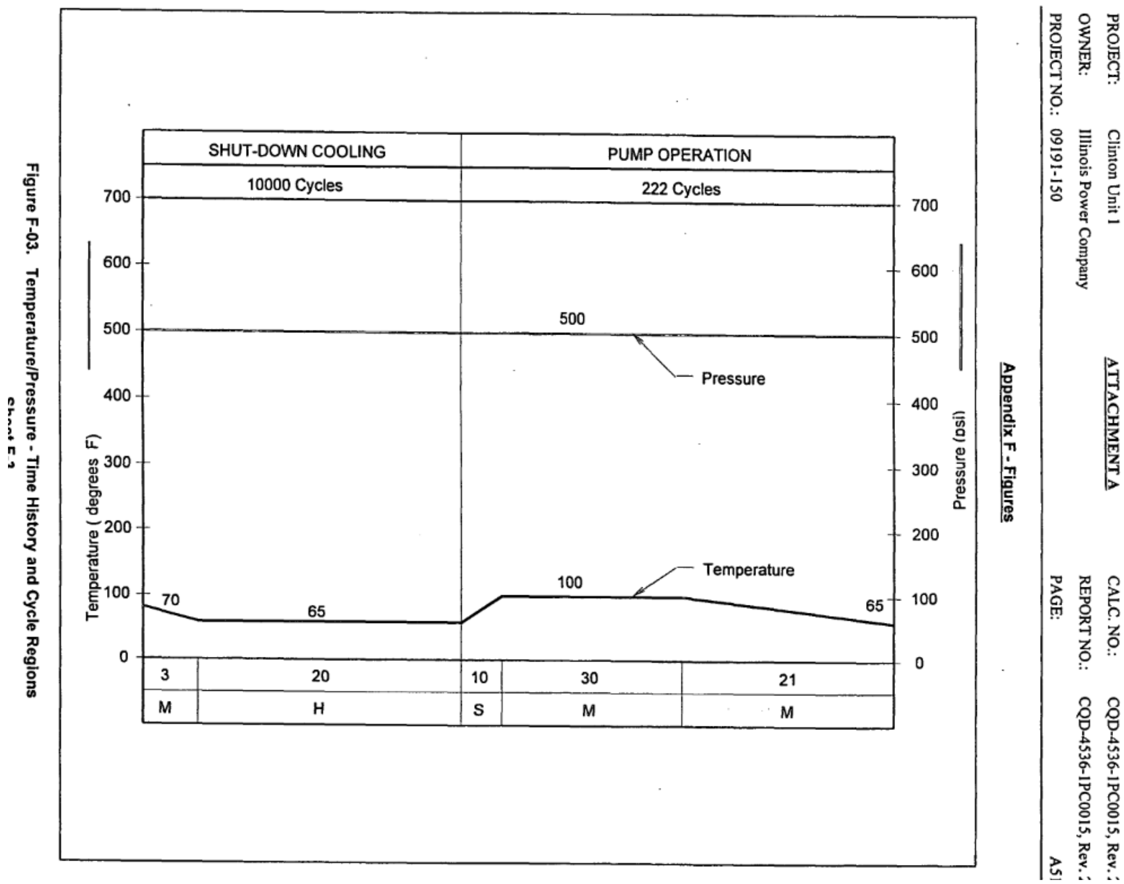

Figure F-03. Temperature/Pressure - Time History and Cycle Regions

The calculation concludes with a CUF of 0.002 based on only the 222 occurrences of the "Pump Operation" transient and does not include a CUF value for 10,000 Shutdown occurrences in which temperature drops 5 degrees F at temperatures below 220 degrees F. Therefore, the Shutdown transient with 10,000 occurrences was not documented in LRA Table 4.6.1-2.

Below is a summary of all other penetrations in LRA Table 4.6.1-2 in which the associated calculations considered transients that specified insignificant temperature changes (e.g. less than 100 degrees F) for which the calculations did not actually calculate an associated CUF value.

- The calculation associated with penetration 1MC-16 considered 10,000 "Shut-Down Cooling" transients where process temperature changes from 70 degrees F to 100 degrees F. This transient was not documented in LRA Table 4.6.1-2.

- The calculations associated with penetrations 1MC-11, 1MC-12, 1MC-13, 1MC-18, 1MC-19, and 1MC-20 considered 10,000 pump "Standby" transients where process temperature remains at 70 degrees F throughout a 10-hour duration. This transient was not documented in LRA Table 4.6.1-2.

- The calculations associated with penetrations 1MC-34 and 1MC-79 considered 10,000 "Pump Standby" transients where process temperature remains at 70 degrees F throughout a 10-hour duration for each occurrence. This transient was not documented in LRA Table 4.6.1-2. Note, these two calculations also considered 108 “SRV Blowdown” transients. This transient is addressed below.

Monitored Transients Described in LRA Table 4.6.1-1

The associated calculations for six of the penetrations in LRA Table 4.6.1-2 assumed monitored transients listed in LRA Table 4.6.1-1. However, the contribution to total fatigue from these monitored transients was very minor. The following detailed explanation illustrates this subset.

The calculation associated with penetration 1MC-69 documents that the only specified transients for this calculation are: 1) 123 “Bolt Up” transients where process temperature remains constant at 70 degrees F, and 2) 657,000 occurrences where process temperature increases from 70 degrees F to 150 degrees F and down to 104 degrees F; 45 times in one day for 40 years, and 3) 130 “Unbolt” transients where temperature remains constant at 70 degrees F. An excerpt of the transient specification from the calculation is shown below.

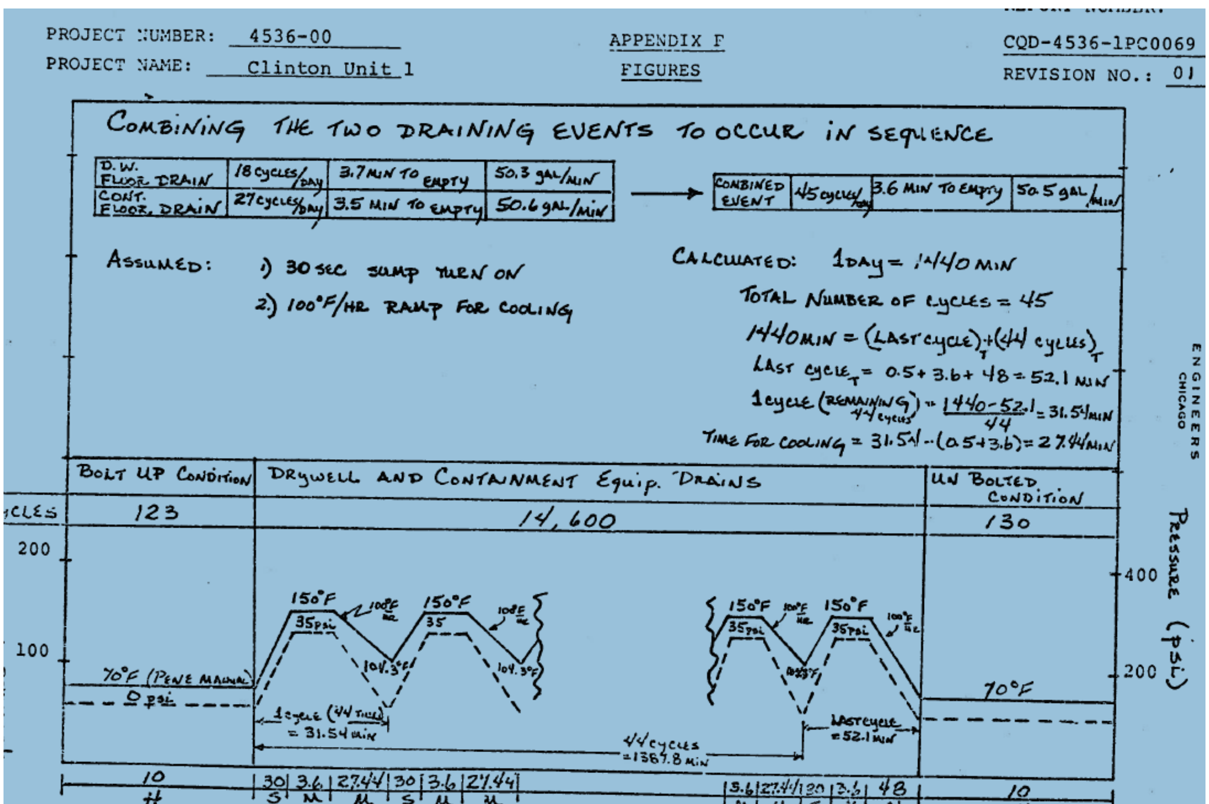

FIGURE F-03. TEMPERATURE/PRESSURE - TIME HISTORY AND CYCLE REGIONS

Although this page specified transients that were called “Bolt up” and “Unbolt” which are documented in LRA Table 4.6.1-1, the specified temperature transients were 10 hour durations of steady state temperature at 70 degrees F and pressure at 0 psig. Therefore, these specified transients have no impact on fatigue for this penetration and these transients were not documented in LRA Table 4.6.1-2. In addition, the calculation associated with penetration 1MC-69 was not included in the tally (4th column) of Table 4.6.1-1 for the “Boltup and “Unbolt” transients.

Below is a summary of all other penetrations in LRA Table 4.6.1-2 in which the associated calculations assumed monitored transients documented in LRA Table 4.6.1-1.

- The calculations associated with penetrations 1MC-70 and 1MC-33 assumed 123 “Boltup” transients where process temperature remains constant at 70 degrees F. The calculations also assumed 130 “Unbolt” transients where process temperature remains constant at 70 degrees F. Therefore, these specified transients have no impact on fatigue for these penetrations and these transients were not documented in LRA Table 4.6.1-2.

- The calculations associated with penetrations 1MC-34 and 1MC-79 assumed 108 SRV Blowdown transients where the suppression pool heats up from 70 degrees F to 140 degrees F back to 70 degrees F and the resulting contribution of these 108 occurrences results in a CUF value of less than 0.005. LRA Table 4.3.1-1 projects three occurrences in 60 years. Therefore, this specified transient has a minor impact on fatigue on these penetrations (a CUF value 0.00014 projected for 60 years). As a result, this transient was not documented in LRA Table 4.6.1-2.

- The calculation associated with penetration 1MC-32, which concluded with a CUF value of 0.00073, assumed 123 “Boltup” transients where process temperature changes from 70 degrees F to 100 degrees F. LRA Table 4.3.1-1 projects 41 occurrences in 60 years. This same calculation also assumed 130 “Unbolt” transients where process temperature remains constant at 70 degrees F. LRA Table 4.3.1-1 projects 40 occurrences in 60 years. Therefore, these specified transients have an insignificant impact on fatigue for this penetration and these transients were not documented in LRA Table 4.6.1-2.

In summary, transients for 10 penetrations that involve insignificant temperature changes are not listed in Table 4.6.1-2. In addition, transients for six penetrations that are included in Table 4.6.1-1 as monitored transients are not listed in Table 4.6.1-2. For all of these exceptions, the transients that are not included in Table 4.6.1-2 have an insignificant or no impact on penetration fatigue calculations.

RCI B.2.1.22-1

Regulatory Basis

Title 10 of the Code of Federal Regulations Section 54.21(a)(3) requires the applicant to demonstrate that the effects of aging for structures and components will be adequately managed so that the intended function will be maintained consistent with the current licensing basis (CLB) for the period of extended operation. As described in the SRP-LR, an applicant may demonstrate compliance with 10 CFR 54.21(a)(3) by referencing the GALL-LR Report when evaluation of the matter in the GALL-LR Report applies to the plant.

Background

Procedure CL-AMPBD-SLI, "Selective Leaching Inspection Sample Basis Document," Revision 0, discussed performing metallurgical analysis of components in place of visual examinations and that the sample size could be reduced by a factor of three (i.e., 6.7 percent or a maximum of 9 components) if the entire material and environment population is evaluated using metallurgical analysis in lieu of visual examinations. During the audit, this procedure was revised to remove the discussion related to reducing sample size if metallurgical analyses are performed.

Request

Confirm that Revision 1 of procedure CL-AMPBD-SLI removed the discussion related to reducing sample size if metallurgical analyses are performed and that the sample sizes at Clinton Power Station will be consistent with GALL LR Report AMP XI.M33, "Selective Leaching."

Constellation Response:

CL-AMPBD-SLI, Revision 1 removed discussion related to reducing sample sizes based on metallurgical analyses. Sample sizes will be consistent with NUREG-1801, XI.M33, "Selective Leaching."