July 1, 2002

MEMORANDUM TO: Geoffrey E. Grant, Director
Division of Reactor Projects
Region III

FROM: Ledyard B. Marsh, Acting Deputy Director ***/RA/***
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

SUBJECT: TASK INTERFACE AGREEMENT (TIA 2001-14) EVALUATION OF LASALLE WATERHAMMER ANALYSIS (TAC NOS. MB3366 AND MB3367)

By memorandum dated November 2, 2001, Region III requested that the Office of Nuclear Reactor Regulation (NRR) determine whether the continuous long term operation of a single train of the residual heat removal (RHR) system in the suppression pool cooling (SPC) mode is within the LaSalle design basis. In addition, Region III requested that NRR review a LaSalle waterhammer analysis to verify that the RHR system will remain operable and/or functional following a loss-of-coolant accident (LOCA) concurrent with a loss of offsite power (LOOP) during operation of the RHR system in the SPC mode. The Region had been discussing this issue with NRR for several months.

NRR has concluded that continuous long term operation of a single train of the RHR system in the SPC mode is within the LaSalle design basis. RHR system analyses which demonstrate that the plant safety systems can withstand a waterhammer event as a consequential failure of a design basis accident (i.e., LOOP/LOCA) are necessary to demonstrate continued compliance with the requirements of 10 CFR 50, Appendix A, General Design Criteria 4, 17, and 35 as part of the design basis. Such analyses ensure that a consequential failure of LOOP/LOCA does not result in a loss of the capability of the RHR system to perform its safety function.

Waterhammer analysis, as part of the RHR system analysis, should demonstrate that the RHR piping will maintain its functional capability and structural integrity for all postulated accidents, and verify that the RHR system post accident function (i.e., Low Pressure Coolant Injection) remains “operational” during any operation in the SPC mode. If a licensee’s analysis for waterhammer does not adequately demonstrate the operability of the RHR system, or that its structural integrity will be maintained, then a single train aligned in the SPC mode should be declared inoperable and its use in that mode restricted by the completion time specified for the applicable Limiting Condition for Operation (LCO) in the plant’s Technical Specifications.

Given that such analyses are within the design basis, the staff has determined that the LaSalle waterhammer analysis contains many simplifying assumptions for which the staff has identified numerous concerns that reflect on the adequacy of the waterhammer evaluation. The staff could not verify that the RHR system will remain operable and/or functional following a LOOP/LOCA during operation in the SPC mode. It is the responsibility of the licensee to demonstrate the operability of the RHR system, and that its structural integrity will be maintained.

The memorandum also indicated that the licensee has commissioned an independent consultant to review the RHR system waterhammer analysis and determine whether this analysis is reasonable to demonstrate system functionality. By letter dated October 10, 2001, Structural Integrity Associates, Inc., reported the results of its evaluation to the licensee and concluded that, while many of the assumptions are reasonable, the analysis employs various assumptions and methodologies that do not consistently follow any approved Code or Regulatory guidance. Although not stated explicitly, the consultant infers that the LaSalle waterhammer analysis contains unquantifiable uncertainties. Therefore, it may not provide reasonable demonstration of the functionality of the RHR system under the postulated waterhammer event.

The staff recommends that the licensee address the specific findings in the attached Safety Evaluation to establish the RHR system operability as discussed in the provisions of Part 9900 of the NRC Inspection Manual. The staff also recommends that the licensee perform an evaluation of the dynamic loading and a detailed non-linear dynamic analysis of the RHR system, subject to ASME Code Section III, Appendix F, criteria. Conversely, the licensee should consider system modifications as a means of avoiding potential line voiding and subsequent waterhammer effects.

The attached Safety Evaluation provides NRR's detailed response to TIA 2001-14.

Because the RHR design basis issue may have generic applicability, this item will be referred to generic issues processes for resolution.

This completes the response to TIA 2001-14 and closes out TAC Nos. MB3366 and MB3367.

Docket Nos. 50-373 and 50-374

Attachment: As stated

cc: B. Platchek, RI
L. Plisco, RII
K. Brockman, RIV

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Docket Nos. 50-373 and 50-374
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO TASK INTERFACE AGREEMENT 2001-14

CONCERNING EVALUATION OF LASALLE WATERHAMMER ANALYSIS

EXELON GENERATION COMPANY, LLC

LASALLE COUNTY STATION, UNITS 1 AND 2

DOCKET NOS. 50-373 AND 50-374

1.0 INTRODUCTION

NRC Information Notice (IN) 87-10, "Potential for Water Hammer During Restart of Residual Heat Removal Pumps," dated February 11, 1987 (Reference 1), and Supplement 1, dated May 15, 1997 (Reference 2), alerted utilities of a potential for waterhammer in boiling water reactor residual heat removal (RHR) systems during a design basis loss-of-coolant accident (LOCA) coincident with a loss of off-site power (LOOP) while the RHR system is in the suppression pool cooling (SPC) mode of operation. The LOOP causes the RHR pumps to stop, which may create voids and column separation in the upper elevations of the RHR piping as a result of water draining back to the suppression pool. When the emergency diesel generators restart the RHR pumps, the water column in the RHR discharge line impacts the stationary water column, thus creating a waterhammer that propagates throughout the RHR system.

At LaSalle County Station (LaSalle), Unit 1, the safety/relief valve (S/RV) leakage rate and lake temperature increased over the 2001 summer months to the point that RHR system operation in the SPC mode was required on a daily basis. The plant operations review committee (PORC) reviewed and approved the manner of operation on June 8, 2001. The technical basis for this decision was documented in Analysis L-002766, "GE NEDC and Continuous Operation of RHR in the Suppression Pool Cooling Mode," Revision 0, dated May 10, 2001 (Reference 3). The PORC considered the S/RV leakage, the valve closure times, the realignment of the system from SPC to low-pressure coolant injection (LPCI) mode, the increased component wear, and continuous operation of RHR in the SPC mode. As a result of the PORC approval, the licensee decided to continuously operate one train of the Unit 1 RHR system in the SPC mode. This operation continued throughout the entire summer and was secured in early September 2001. Since no change to the facility as described in the LaSalle Updated Final Safety Analysis Report (UFSAR) was identified, the licensee did not perform a formal review against the requirements of 10 CFR 50.59.

By memorandum dated November 2, 2001 (Reference 4), Region III requested that NRR determine whether the continuous long term operation of a single train of the RHR system in the SPC mode is within the LaSalle design basis. In addition, Region III requested that NRR review a LaSalle waterhammer analysis to verify that the RHR system will remain operable

and/or functional following a LOCA concurrent with a LOOP during operation of the RHR system in the SPC mode. Region III had been discussing this issue with NRR for several months.

2.0 REGULATORY EVALUATION

NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants - LWR Edition," dated July 1981, provides the guidance for staff reviewers performing safety reviews. The Standard Review Plan (SRP) sections are keyed to the format identified in Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants - LWR Edition," and are intended to comply with the regulatory requirements of 10 CFR 50, Appendix A.

SRP Section 3.6.2 is intended to comply with the requirements of General Design Criteria (GDC) 4, "Environmental and Missile Design Bases." GDC 4 requires that structures, systems, and components important to safety shall be designed to accommodate the effects of postulated accidents, including appropriate protection against dynamic and environmental effects of postulated pipe ruptures. However, dynamic effects associated with pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.

SRP 3.6.2 Branch Technical Position (BTP) MEB 3-1, "Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment," provides that "through-wall leakage cracks instead of breaks may be postulated in the piping of those fluid systems that qualify as high-energy fluid systems for only short operational periods but qualify as moderate-energy fluid systems for the major operational period." High-energy systems include those systems where either of the following conditions are met: (a) the maximum operating temperature exceeds 200°F, and (b) the maximum operating pressure exceeds 275 psig.

GDC 17, "Electric Power Systems," requires that an onsite electrical power system and an offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety. Provisions shall be included to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies.

GDC 35, "Emergency Core Cooling," requires that a system to provide abundant emergency core cooling shall be provided. Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure. If the LOOP is more limiting, the licensee is required to consider a LOCA concurrent with the LOOP (i.e., LOOP/LOCA event).

3.0 TECHNICAL EVALUATION

3.1 LaSalle Design Basis

LaSalle's containment heat removal system is described in Section 6.2.2 of the UFSAR. Part of that description includes operation of the RHR system in the SPC mode. However, the number of cycles and duration of operation of the RHR system in the SPC mode is not explicitly described here in the UFSAR or elsewhere in the licensing basis. Also, the LaSalle Technical Specifications (TS) bases, Section B.3.6.2.3, "Residual Heat Removal (RHR) Suppression Pool Cooling," background, states in part:

> The heat removal capability of one RHR pump in one subsystem is sufficient to meet the overall DBA [design basis accident] pool cooling requirement to limit peak pool temperature to 208 degrees F for loss of coolant accidents (LOCAs) and transient events such as a turbine trip or stuck open safety/relief valve (S/RV). S/RV leakage and Reactor Core Isolation Cooling System testing increase suppression pool temperature more slowly. The RHR Suppression Pool Cooling System is also used to lower the suppression pool water bulk temperature following such events.

While it is clear from the TS Bases that the NRC staff acknowledged RHR operation in the SPC mode for S/RV leakage, it was the staff's expectation that the S/RVs would be well maintained such that any leakage would be minor and the use of SPC would be infrequent and of short duration. Therefore, the staff did not consider the continuous long-term operation of the RHR system in the SPC mode to be part of the original design basis.

Although extended use of RHR in the SPC mode was not considered part of the original design basis, licensees must still meet the requirements of GDC 4, 17, and 35, which state that a nuclear power plant must withstand a LOCA with unavailability of either onsite or offsite power, and any consequential failures of the event. Therefore, potential waterhammer in the RHR system resulting in either through-wall leakage cracks or pipe ruptures induced by a LOOP/LOCA is a design basis event.

In Reference 1, the staff informed boiling water reactor licensees of the potential for a waterhammer in the RHR system during a design basis LOOP/LOCA if one or more RHR loops are in the SPC mode. In response, LaSalle documented that due to a "zero leakage" program for S/RVs and due to the specific configuration of plant equipment, the waterhammer scenario described in IN 87-10 was unlikely. As a result of NRC concerns that IN 87-10 had not been adequately addressed, Sargent and Lundy (S&L) performed a LaSalle waterhammer analysis EMD-067982, "Evaluation of Potential Water Hammer in Residual Heat Removal System," Revision 0, dated February 18, 1994 (Reference 5). This analysis concluded that although a waterhammer could occur, the RHR system would maintain its pressure boundary integrity, structural stability, and functional capability during the waterhammer event. However, NRC inspectors noted that plastic deformation and ovalization of system piping as well as snubber failure were also predicted. These results were subsequently documented in the LaSalle UFSAR.

In December 1995, General Electric (GE) issued Report NEDC-32513, "Suppression Pool Cooling and Water Hammer" (Reference 6), documenting their review of the generic

waterhammer issue. This report indicated that operation of the RHR system in SPC mode was expected to be an infrequent occurrence. As a result, the original design basis and supporting analysis assumed LPCI initiation from the standby configuration, not from the SPC mode. This report also noted that, “the expected operating time in the SPC mode was estimated to be sufficiently low to satisfy the NRC position on exemptions for ‘short operational periods’ (e.g., NUREG-0800, Section 3.6.2 on pipe breaks).” GE further stated that "a mechanistic consequence such as waterhammer as a result of a LOCA concurrent with a LOOP was not an original licensing requirement and was neither intended nor included in the original design and in the licensing review process." However, the NRC exemption per GDC 4 is only for dynamic effects of postulated pipe rupture (due to low probability) and does not imply an exemption from analyzing postulated design basis accidents and their consequential failures.

On May 15, 1997, the staff issued Supplement 1 to IN 87-10. The supplement alerted licensees to the continuing potential for waterhammer in the RHR system during a LOOP/LOCA if aligned in the SPC mode. The supplement also addressed the increased use of RHR pumps in the SPC mode due to leaking S/RVs, and it specifically stated that this greater operating time may be more than that assumed in the original design basis. On October 12, 1997, the staff issued TIA 96-0389, "Quad Cities, Unit 1 and 2, Regarding NEDC-32523 Applicability to RHR Water Hammer Potential," which reviewed the conclusions in the 1995 GE report. Based on these, NRC inspectors reviewed LaSalle's 1994 waterhammer analysis and concluded that it was not required as long as RHR operation in SPC mode was limited to the short operational periods assumed in the design basis.

SRP 3.6.2, BTP MEB 3-1, states that, “an operational period is considered ‘short’ if the fraction of time that the system operates within the pressure-temperature conditions specified for high-energy fluid systems is about 2 percent of the time that the system operates as a moderate-energy fluid system (e.g., such as the decay heat removal system qualify as moderate-energy fluid systems ...).” In the 1995 GE report, “short operational period” was also defined as 2 percent of the time, referring to SRP 3.6.2. The normal operating time of the SPC mode is estimated to be less than 2 hours for every reactor core isolation cooling (RCIC) system surveillance test, usually every 92 days. While the phrase “short duration” is not explicitly characterized in the LaSalle UFSAR, other documents make clear that “short duration” was the GE RHR design basis and that the licensee was aware of, and adhering to, this understanding in its operation prior to its June 2001 decision to operate RHR continuously in SPC mode.

LaSalle decided to operate in the SPC mode of operation for long periods of time, from June 2001 to September 2001, which clearly exceeded the “short” duration (defined as 2 percent of the time) of operation. In June 2001, LaSalle commenced continuous operation of one train of RHR in the SPC mode due to S/RV leakage and summer temperatures. Even though there was no change in the plant operating procedures or the physical facility as described in the UFSAR, the frequency of SPC operation was significantly increased beyond what was assumed in the design basis. The staff has determined that this was a change in the design and performance requirements of a system described in the UFSAR, which means a "change in the facility" as defined in 10 CFR 50.59.

The "short operational period" definition referred to in the 1995 GE report is only applicable to operation of the RHR system in the Shutdown Cooling (SDC) mode of operation. The 2 percent criteria only defines the low probability threshold for piping rupture while in a high-energy fluid system mode. In SDC mode, the RHR system will be operating above 200 °F and 275 psig for

limited periods and would qualify as a high-energy fluid system subject to the 2 percent criteria. The flow paths in the RHR system that are used for SDC and SPC modes share common piping segments. Therefore, the RHR system needs to be analyzed to accommodate the effects of waterhammer during both of these modes, as well as all other modes of operation.

As previously discussed, the licensee is required to show that it has analyses which demonstrate that the plant safety systems can withstand a waterhammer as a consequential failure of a required design basis accident (i.e., LOOP/LOCA). Such analyses are necessary to demonstrate continued compliance with the requirements of GDC 4, 17, and 35 as part of the design basis, and ensure that a single failure does not result in a loss of the capability of the RHR system to perform its safety function. The required RHR system analysis would be bounding for all modes of operation at all times, regardless of which specific mode the system is aligned in at any particular time. The amount of time spent in a medium-energy fluid system mode of operation is not defined in SRP 3.6.2.

At the time of licensing, the waterhammer concern was not recognized and thus no action was required or taken. However, based on IN 87-10 (as supplemented) which identified this concern, the licensee should have verified their analysis of RHR operation in the SPC mode, and should have reviewed the change in frequency of SPC operation in accordance with the 10 CFR 50.59 change process prior to deliberately undertaking long term operation in that mode in June 2001. Since the probability of a waterhammer event increases as the amount of time the RHR system is operated in the SPC mode increases, the likelihood of damage to the system increases. Therefore, the staff does not recommend RHR operation in the SPC mode for more than a “short” period of time.

In summary, the staff concludes that continuous long term operation of a single train of the RHR system in the SPC mode is within the LaSalle design basis. RHR system analyses which demonstrate that the plant safety systems can withstand a waterhammer event as a consequential failure of a design basis accident (i.e., LOOP/LOCA) are necessary to demonstrate continued compliance with the requirements of GDC 4, 17, and 35 as part of the design basis. Such analyses ensure that a consequential failure of LOOP/LOCA does not result in a loss of the capability of the RHR system to perform its safety function.

Waterhammer analysis,as part of the RHR system analysis, should demonstrate that the RHR piping will maintain its functional capability and structural integrity for all postulated accidents, and verify that the RHR system post accident function (i.e., LPCI) remains “operational” during any operation in the SPC mode. Although SPC operation is bounded by RHR system analysis and its frequency is not restricted, unless otherwise specified in the licensing basis, the staff expects that use of SPC during normal operation would be of short duration and that any significant increase in frequency assumed in the design basis be reviewed in accordance with the 10 CFR 50.59 change process.

If a licensee’s analysis for waterhammer does not adequately demonstrate the operability of the RHR system, or that its structural integrity will be maintained, then a single train aligned in the SPC mode should be declared inoperable and its use in that mode restricted by the completion time specified for the applicable Limiting Condition for Operation (LCO) in the plant’s Technical Specifications.

3.2 LaSalle RHR System Waterhammer Analysis

The hydraulic analysis performed by S&L determined the transient waterhammer forces in the RHR system, using the S&L hydraulic analysis program HYTRAN. This analysis calculated the hydraulic force time histories at various locations of the piping, caused by the propagation and reflection of the waterhammer pulses. These forces act at locations in the piping where there is a change in fluid momentum direction, such as at elbows and tees. The structural transient analysis was performed using the S&L program PIPSYS. This is a linear-elastic program for piping analysis that has the stated capability of performing dynamic time-history analysis.

The RHR system consists of a number of sub-systems. A PIPSYS transient analysis of the RHR system was performed, using the HYTRAN generated hydraulic force-time histories as inputs. The sub-system 1R-H24 was determined as the highest loaded system, and its safety evaluation was reported in S&L Report EMD-067982 (Reference 5).

The staff reviewed specifically Attachment 8.4 to the report, “Piping Calculations to Justify the Pressure Boundary Integrity, Functionality and Structural Stability of Subsystem 1RH-24," that forms the basis for the S&L conclusions regarding operability of the RHR system. The following is an evaluation of selected sections of Attachment 8.4:

Section 2: Assumptions

This section of EMD-067982 lists the assumptions on which the analysis is based.

2.1 “High damping values could be used in the dynamic force-time history analysis, since the response is expected to exceed yield strength deeply into the plastic zone”.

S&L specified damping as 15 percent of critical damping in the elastic piping analysis, and adopted this value from data described in Volume 2 of the Electric Power Research Institute (EPRI) Piping and Fitting Dynamic Reliability Program Report, dated December 1989. S&L justified this damping value because of expected considerable plastic deformation. The value of 15 percent equivalent damping was determined from a technique used by EPRI whereby dynamic elastic analysis of the components was performed, subjected to broadened damped seismic response spectra. The calculations were performed with various values of assumed damping coefficients until the highest elastically calculated moment matched the measured test (inelastic) moment. This approach is valid only for the particular tested configuration, subjected to the particular seismic loading input. It is not valid for any other configuration or loading history, and generally not valid where inelastic deformation occurs. The NRC staff has not accepted this procedure for determining damping nor the damping values obtained as a result of its application. In addition, the considerable plastic deformation that is expected in the analysis makes the elastic analysis using PIPSYS questionable. The high values of damping used in the elastic analysis also lead to underestimation of the displacements and strains in the affected piping.

The staff concludes that the licensee did not provide adequate basis to justify the exceedingly high damping value of 15 percent in their analysis of the RHR system for waterhammer. Although not endorsed by the staff, the 5 percent damping value specified in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code) Section III, Appendix N, for the dynamic analysis of elastic piping structures is considerably lower than the value used by the licensee.

2.2 “Plasticity Strain Hardening (including cyclic and strain rate effects) can be conservatively set to 10% of the elastic modulus.”

This is the modulus of elasticity used in the PIPSYS analysis to calculate nominal elastic bending stresses and plastic deformations.

It is not clear what is meant here by “strain-hardening.” Ordinarily, this term refers to the slope of a bi-linear stress-strain curve in the work-hardening range if an elastic-plastic analysis were used. For the stated RHR piping material, SA-106 Gr. B, this value is in the order of 127 ksi. As used here, it is the modulus of elasticity that was used in a PIPSYS linear-elastic analysis to eventually calculate plastic ratchet strain increments with the “modified Bree model” in Assumption 2.7 below. No justification is provided regarding the conservatism in this assumption, since the piping may undergo large deformations. Therefore, the validity of the PIPSYS analysis is questionable.

2.3 “Piping components (elements) which exceed Level D service limit, due to the dynamic transient loading, can be modeled with soft material properties to assess the subsystem stability under weight loading.”

The stability analysis of any system is a non-linear problem, whether in the elastic or plastic domain. It depends on the changes in geometry that the structure experiences under loading and the interaction of the loads and the geometry. Simply changing a material property alone in some components is insufficient to determine stable or unstable states. Therefore, this assumption is not valid for determining stability from a linear-elastic analysis.

2.4 “The number of equivalent cycles at the maximum response magnitude is 5 cycles.”

S&L is referring to structural vibratory cycles. No supporting basis for this assumption is apparent since stress time-histories have not been provided, and its validity cannot be ascertained. This assumption may also be non-conservative since it depends on the assumed damping in the piping analysis.

2.5 “The elastic-plastic strain distribution within the elbow body is identical to that of the elastic strain.”

This assumption states that the elastic-plastic strains in an elbow can be determined from an elastic analysis with a reduced modulus. There is no basis for this assumption in elastic-plastic theory. It is used with Methodology 3.2 below and Assumption 2.7 in the “modified Bree model” to calculate the elastic peak strain in an elbow. This assumption is not accurate, since the elastic-plastic strain distribution is not the same as the elastic distribution in the “Bree model,” or the “modified Bree model,” as used in this report.

2.6 “The maximum steady state internal pressure the piping system would be subjected to in the suppression pool cooling mode is 500 psig.”

The staff cannot verify the basis for this assumption since no reference is provided in the design documentation. This could be misinterpreted as qualifying the RHR system in SPC mode as a high-energy fluid system, which is unlikely since system pressure is expected to be

maintained well below 275 psig during normal operation. The maximum RHR system pressure would be 500 psig based on interlocks for LPCI injection and system relief valves, but this would only occur for a limited period during a LOCA blowdown.

2.7 "Modified Bree model with Sy=Sf=(Sy+Su)/2 can be utilized to assess the maximum cumulative ratchet strain."

Reference 7 describes the elastic-plastic analysis of a cylindrical thin-walled tube subjected to constant internal pressure and cyclic radial thermal through-wall gradients. This analysis calculates cyclic plasticity or a form of incremental collapse of the tube called ratcheting. The "Bree model" consists of modeling the tube by a straight bar of unit width and tube wall thickness with its ends prevented from bending, subjected to the steady hoop stress in the tube. This is possible if the steady axial stress in the tube due to pressure is disregarded. The elastic-plastic analysis of this model permits the calculation of incremental hoop ratchet strains in closed form for a tube of elastic, perfectly-plastic material under these particular loading conditions. Whether plastic cycling or ratcheting occurs depends on the magnitude of the hoop stress and the maximum thermal stress. The stress in the wall does not exceed the yield stress of the material. The "modified Bree model" used in the report consists in adapting the "Bree model" to elbows and tees subjected to internal pressure and cyclic vibratory loading, caused by ovalization of the cross-sections due to mechanical moment loading. However, the analysis of the "Bree model" is based on thermal loading that is cyclically applied and removed. Under waterhammer vibratory loading, the bending moment in the elbow wall may alternate between positive and negative values. When combined with internal pressure, it is thus possible to have plastic cycling, or combined ratcheting and plastic cycling. No justification was presented that the "Bree model" and analysis are also applicable to this type of loading condition, and no justification was presented that the "Bree model" and analysis are also applicable to elbows and tees. S&L used the maximum nominal bending stress range (calculated from PIPSYS under Assumption 2.2) in the "modified Bree model" to address alternating moment loading. The validity and conservatism of this approach cannot be ascertained, since the "Bree model" does not reflect alternating thermal loading.

A "flow stress," defined as the average of the pipe material yield and ultimate stresses, was also used as a yield stress with the "modified Bree model." The "Bree model" is based on the material yield stress and modulus of elasticity, not on a flow stress. For an elastic-work-hardening material, the calculation of the incremental ratchet strains is far more complex, as shown in Reference 7. The reason for using the flow stress apparently was to avoid these complex calculations. However, taking the average of the yield and ultimate stresses is valid only for problems where the deformations may be expected to reach the ultimate strain of the material. Acceptance Criteria 4.1 below indicates that the strain should not exceed 5 percent. Therefore, the flow stress should be based on the average of the yield stress and the stress corresponding to a strain of 0.05. For the pipe material, this happens to be close to the yield stress.

2.8 "For Welding Tee, secondary bending strains can be conservatively estimated as e_b = 0.675*maximum apparent primary strain, or e_b = 0.403*maximum apparent Primary Plus Secondary Strain."

No justification is presented for this assumption or for its conservatism. The concepts of primary strains, or primary plus secondary strains, are not defined in ASME Code Section III.

The staff has not been able to verify the basis for this assumption. The licensee should provide adequate basis to demonstrate the conservatism of this assumption.

2.9 "For Welding Tee, Local Primary membrane stress due to pressure can be estimated as 1.5 times the nominal pressure stress."

No justification for this assumption is presented. The staff cannot identify or verify the basis or conservatism for this assumption.

Section 3.0: Methodology

This section of EMD-067982 addresses the analytical methods used to assess stresses and strains.

3.1 A linear-elastic dynamic analysis was performed using PIPSYS with $E = 30*10^6$ psi and 15 percent damping.

This analysis determined the locations where the Level D service limit was exceeded. Using this damping value, assumed uniform over the entire piping system, underestimates the stress intensity at all locations. As stated under Assumption 2.1, 15 percent damping does not conform with the staff regulatory position or current industry practice. The results from the PIPSYS analysis were also used in an ASME Code Section III Class 1 analysis to determine the fatigue usage due to the stress cycling. It is not clear that loading included stress cycling as a result of the passing of the pressure pulses. The fatigue usage is most likely underestimated as a result of using the high damping in the analysis.

3.2 The PIPSYS linear-elastic analysis of the system was repeated with 15 percent damping and $E = 3*10^6$ psi. This value for E was chosen to represent 10 percent strain-hardening, as stated in Assumption 2.2.

The purpose of this calculation was to calculate "membrane plus bending" stresses that would be used with the "modified Bree model" analysis to determine ratcheting strains. Using a lower value of E and a high damping value with a linear-elastic calculation is a convenient but unsound artifice for evaluating the plastic deformation of the system. It implies the entire system is deforming inelastically, which is not correct. Plastic deformation in a piping system occurs at localized, high stress locations where "hinges" may form. Between these "hinges," the system remains elastic. Assuming that the entire system deforms inelastically may underestimate the largest localized plastic deformations.

As stated above under Assumption 2.2, there is no basis for using a lower value of E in a linear-elastic (or non-linear elastic) piping analysis to evaluate the plastic behavior of the piping system. This procedure is inconsistent with accepted general principles and methods of plastic analysis of solids and structures. (Methods to solve linear and non-linear material and geometric dynamic problems are described in ASME Code Section III, Appendix N.) In addition, the potentially large changes in geometry invalidates the elastic analysis, which is based on small deformations and small displacements.

3.3 A PIPSYS linear-elastic analysis of the system was performed with E = 10^5 psi and Poisson's ratio = 0.5 to study the inelastic stability of the system under dead weight only.

It is not clear how a Poisson ratio of 0.5 is introduced in the elastic calculation, since this implies incompressibility of the material. Stability of elastic or inelastically deformed systems are non-linear problems and can be determined only by the methods of stability of structures, including limit analysis. In this case, the stability of the structure should be determined under the dead weight and the transient loads acting simultaneously. Therefore, this analysis is not meaningful.

Section 4.0 Acceptance Criteria

4.1 "Maximum membrane plus bending strain amplitude due to the fluid transient load should not exceed 5%."

This criterion is acceptable if an elastic-plastic large deformation analysis is performed. It is not acceptable for strains calculated on a linear-elastic basis, such as in PIPSYS. (The minimum yield strain for the pipe material is 0.00107 in/in. A strain of 0.05 is 47 times as large as the yield strain. It is about 23 percent of the ultimate strain of the material.)

4.2 "Maximum accumulated ratchet strain should be less than 5%."

This criterion is acceptable.

4.3 "Maximum pipe cross-section ovality should not cause an area reduction more than 10%."

This criterion is acceptable.

4.4 "Estimated fatigue usage factor for the fluid transient piping response utilizing the provisions of Class-1 analysis should not be excessive."

The term "excessive" is not defined. This criterion may be acceptable, provided the ASME Code Section III fatigue usage limit of 1.0 is not exceeded.

4.5 "Stability of the piping system with plastic hinges, at the overloaded locations, subjected to its own weight can be justified if the maximum weight loading deflection in any direction is less than 5.0" with no excessive loads on restraints."

This criterion to demonstrate system stability is meaningless. The calculation uses the undeformed configuration, and does not consider the interaction with the transient loads. In addition, loads on the restraints should not exceed the limiting loads calculated using the criteria of ASME Code Section III, Appendix F.

4.6 "Local strain level in the pipe component should not exceed the critical buckling strain."

This criterion is acceptable.

Section 5.0 "Physical Data"

This section provides the yield and ultimate strengths of the pipe material, SA-106, Gr. B, at the operating temperature, and calculates the "flow" stress that is used in the "Bree" analysis. These values appear correct.

Section 6.0 "Calculations"

6.1 General Observations

The highest stress point reported from the analysis with $E=30*10^6$ was at the elbow located at L25. Higher stress points were not reported, such as at the branch connection L3, for unverifiable reasons.

The report states that the "apparent fatigue usage per single maximum flow transient response cycle is 0.3. This translates to a usage factor = 1.5 for the estimated five equivalent dynamic cycles. Compared with the fatigue margin of 20, a usage factor = 1.5 would be acceptable, since the ASME Code Section III would allow up to 25 stress cycles which would exceed Sa @ 10^6 to be excluded from fatigue consideration for Service Level C condition (NB-3113(b))." This justification is unclear, and the calculation of the 0.3 fatigue usage factor per dynamic cycle was not reported and cannot be evaluated or verified.

The ASME Code Section III design fatigue curves are based on fatigue curves that were obtained from strain-controlled laboratory tests on small polished samples at room temperature in air. The design fatigue curves were calculated from these tests by decreasing the best-fit curves to the laboratory test data by a factor of 2 on strain or 20 on cycles, whichever was more conservative, at each point on the best-fit curve. The ASME Code Section III Stress Criteria document indicates that these factors were intended to account for the differences and uncertainties in relating the fatigue lives of laboratory test specimens to those of actual reactor components, in actual reactor environments. Paragraph NB-3121 of ASME Code Section III, Subsection NB, also states that the data on which the fatigue design curves are based did not include tests in the presence of corrosive environments that might accelerate fatigue failure. As stated in Reference 8, the factors of 2 and 20 are not safety margins but rather conversion factors that were applied to experimental data to obtain reasonable estimates of the lives of actual reactor components. Therefore, the assertion by S&L is not acceptable. A fatigue usage factor > 1.0 implies a crack has initiated and is in the process of propagating. Based on the S&L calculation, the staff concludes that there is a significant probability that cracking will initiate and propagate in the highest loaded component.

6.2 Strain Evaluation

This section calculates the potential ratcheting in the highest stressed components resulting from the waterhammer loading, using the "modified Bree model." As stated above, the "modified Bree model" consists of replacing the following: (1) a nominal thermal stress term in the "Bree model" by a nominal bending stress range, (2) the yield stress with the flow stress, and (3) the modulus of elasticity by the reduced modulus. No justification was presented that the "Bree model," or the "modified Bree model," are applicable to elbows and tees subjected to alternating deformation-controlled bending moments.

The highest nominal elastic elbow and tee bending stresses were determined from the PIPSYS analysis that used the reduced modulus of $E = 3.0*10^6$ psi and 15 percent damping. These stresses were used to determine incremental ratchet strains from the “modified Bree model” analysis. Since the analysis is based on 15 percent damping, these stresses and strains are most likely underestimated.

The nominal bending stress ranges were converted to strains by dividing these stresses by the reduced modulus, except for a yield strain term, called the “associated apparent elastic strain.” Based on the stresses calculated from the elastic analysis with the reduced modulus, S&L calculated “nominal” and “maximum membrane + bending” strains in three high stressed components, two elbows and a tee. To calculate the “nominal” strain, the “nominal” axial pipe stress was first determined from the elbow bending stress (calculated from Equation 9 of the ASME Code Section III, Subsection NB), divided by the Code B_2 factor. (For the 12" L. R. Sch. 40 elbow at location L25, this factor was reported as 4.4. The staff calculated this value as 3.908. There is thus an unexplained discrepancy.) The “nominal” axial pipe stress was multiplied by the Code C_2 factor to determine the maximum elastic bending stress. The maximum strain, called the “maximum membrane + bending” strain, was obtained by dividing the maximum bending stress by the reduced modulus. The “bending strain amplitude” was calculated by subtracting the “nominal” axial strain from the “maximum membrane + bending” strain that is mostly in the hoop direction. These strains do not even occur at the same location and this step is without basis. The "associated apparent elastic strain" was calculated by dividing the “flow” stress term in the “modified Bree model” formula by the material modulus of elasticity. The “modified Bree model” thus used two different modulii of elasticity, which is inconsistent with the “Bree model” and has no basis.

For the welding tee, the "Bree" analysis is based on unverifiable assumptions (e.g., Assumption 2.8).

Once the incremental ratchet strains were calculated, the cumulative ratchet strains (CRS) were calculated by multiplying the incremental ratchet strains by five dynamic cycles, per Assumption 2.4 above. As stated earlier, it is not clear if this refers to five hydraulic pulses or if these are vibratory mechanical cycles.

S&L reported the calculated CRS for the following components:

1) 12" L. R. Elbow @ L25, CRS = 0.04984 in/in

2) 16" L. R. Elbow @ L125, CRS = 0.052 in/in

3) 16" Welding Tee @ L110, CRS = 0.0528 in/in

For two of the three components, the CRS exceed the ratchet strain criterion stated in Assumption 4.2 above, and therefore fail to meet the S&L imposed criterion. Furthermore, these cumulative ratchet strains are most likely underestimated, as compared to those calculated using an elastic-plastic analysis, since they are based on elastic analysis with high damping.

The purpose of the calculation with the reduced modulus is not clear. The maximum elastic stress for the elbow at L25 was reported as 130 ksi, based on the material modulus of $30*10^3$

ksi. Using these values and the material yield stress of 32 ksi, and following the S&L procedure (except that the same modulus was used throughout), the cumulative ratchet strain was calculated as 0.0212 in/in, lower than shown above. The high stresses at the other components were not reported. On the same basis as for the elbow, it is possible that the CRS for these components was higher than those listed above. However, as also stated above, all these CRS may be underestimated, since they are based on an elastic analysis with high damping.

6.3 Functional Capability Assessment

This section calculates the flow area reduction as a result of the ovalization of the elbows under ratcheting. It is based on the calculated “maximum membrane + bending” strain. However, unless a more detailed analysis is performed, it should be based on the accumulated ratchet strain. The reduction in flow area may be greater than that stated in the report. The licensee should evaluate the reduction in area by including the cumulative ratcheting strain.

6.4 Local Critical Buckling Assessment

The S&L analysis is based on calculating the elastic axial critical buckling strain in a thin walled cylinder or panel from cases 13 and 15 in Roark (Reference 9). This strain is compared to the “maximum membrane + bending” strain, which is a hoop strain. Therefore, there is no basis for this comparison. The critical buckling assessment should be based on the collapse moments for an elbow or tee.

6.5 Global Structural Stability

The stability of the system was evaluated by a linear-elastic analysis under dead weight, where the modulus of elasticity of the high stressed members was reduced to 100,000, to represent plastic "hinges." As stated in Assumption 4.5 above, the criterion adopted in this report for system stability is that the maximum deflection under weight not exceed 5.0 inches. An ordinary criterion for stability is whether the structure will collapse. To determine structural collapse requires that there be interaction between the external loading and the deformed geometry of the structure. A maximum displacement criterion is acceptable provided it is determined from an analysis that considers load-displacement interaction. The S&L approach does not provide assurance whether the structure will remain stable or not.

The tables titled “Support Load Comparison from PIPSYS Analysis (Run ID S15ALL)” and “Support Load Comparison from PIPSYS Analysis (Run ID S15PCT)” indicate that the loads in the supports are all listed as positive. This means that either the supports are all under tension type loading, or compression type loading has been ignored. The footnotes to these tables show that the capacity of the supports was taken as 2*Faulted Capacity for snubbers, and 4*Faulted Capacity or 10*Service Level A/B Capacity for rigid support and struts. As stated in ASME Code Section III, Paragraph NCA-2142.2(b)(4), the limits of ASME Code Section III, Appendix F (Appendix F), permit gross general deformations with some consequent loss of dimensional stability and damage requiring repair, which may require removal of the component from service. Therefore, the stated capacities for the supports exceed the limiting capacities of Appendix F by considerable margins and are not acceptable.

Based on the above discussion, the following is a summary of the staff evaluation of the LaSalle RHR system waterhammer analysis:

- The licensee did not provide sufficient information regarding the hydraulic transient analysis using HYTRAN. Therefore, the staff could not evaluate its adequacy. The staff examined the diagrams of transient hydraulic forces acting on the piping that were included in EMD-067982, but cannot conclude that the forces have been acceptably calculated.

- Part 9900 of the NRC Inspection Manual (Reference 10), Section 6.13, "Piping and Pipe Support Requirements," specifies that operability of non-conforming piping can be demonstrated by meeting the Appendix F limits and criteria for Level D Service loading. Paragraph NCA-2142.2(b)(4) of ASME Code Section III states that "these limits permit gross general deformations with some consequent loss of dimensional stability and damage requiring repair, which may require removal of the component from service." Section F-1200(a) of Appendix F also states that these limits "are intended to assure that violation of the pressure retaining boundary will not occur, but are not intended to assure the operability of the components during or following the specified events." Nevertheless, the NRC has adopted these rules to permit demonstration of operability and functional capability of piping and piping systems at any time during the operating cycle until the first upcoming plant outage, when the piping can be inspected and repaired or replaced, as appropriate.

Reference 4 has also indicated that the licensee commissioned an independent consultant, Structural Integrity Associates, Inc. (SIA), to review EMD-067982 and determine whether the analysis is reasonable to demonstrate system functionality. By letter dated October 10, 2001 (Reference 11), SIA reported the results of its evaluation to the licensee. SIA reviewed the analysis, along with Regulatory, Code and industry guidance, to assess the validity of the approach used, and to determine the potential for meeting Appendix F requirements. Based on its assessment, SIA stated that while many of the assumptions were considered as reasonable, the report did not provide sufficient justification for the approach used in the analysis. SIA concluded that the S&L piping structural analysis employed various assumptions and methodologies that did not consistently follow any approved Code or Regulatory guidance, a conclusion similar to that reached by the staff. SIA recommended an approach to demonstrate compliance with Appendix F, consisting of adopting the Appendix F criteria as the acceptance criteria, perform time-history inelastic finite element analysis of the RHR system using actual material stress-strain curves, and limit the dynamic analysis damping to 5%. (SIA did not recommend a re-evaluation of the hydraulic time-history forcing functions.) As this is an extensive and highly costly analysis, and no assurance can be provided that the criteria of Appendix F would be met, SIA recommended, as an alternative, that the licensee consider installation of check valves as one means of preventing line voiding and subsequent waterhammer loading.

The staff concludes that the S&L analysis of the RHR system at LaSalle, as reported in EMD-067982, contains many simplifying assumptions for which the staff has identified several concerns that reflect on the adequacy of the waterhammer evaluation to demonstrate the operability of the RHR system, or that its structural integrity will be maintained. Therefore, the staff recommends that the licensee should address the specific findings in the above staff assessment to establish the RHR system operability as discussed in the provisions of Part 9900

of the NRC Inspection Manual. The staff also recommends that the licensee perform an independent evaluation of the dynamic loading and a detailed non-linear dynamic analysis of the RHR system, subject to Appendix F criteria. Conversely, the licensee should consider system modifications as a means of avoiding the line voiding and the subsequent waterhammer event.

4.0 CONCLUSION

The NRC staff has determined that continuous long term operation of a single train of the RHR system in the SPC mode is within the LaSalle design basis. RHR system analyses which demonstrate that the plant safety systems can withstand a waterhammer event as a consequential failure of a required design basis accident (i.e., LOOP/LOCA) are necessary to demonstrate continued compliance with the requirements of GDC 4, 17, and 35 as part of the design basis. Such analyses ensure that a consequential failure of LOOP/LOCA does not result in a loss of the capability of the RHR system to perform its safety function.

Waterhammer analysis, as part of the RHR system analysis, should demonstrate that the RHR piping will maintain its functional capability and structural integrity for all postulated accidents, and verify that the RHR system post accident function (i.e., LPCI) remains “operational” during any operation in the SPC mode. Although SPC operation is bounded by RHR system analysis and its frequency is not restricted, unless otherwise specified in the licensing basis, the staff expects that use of SPC during normal operation would be of short duration and that any significant increase in frequency be reviewed in accordance with the 10 CFR 50.59 change process.

If a licensee’s analysis for waterhammer does not adequately demonstrate the operability of the RHR system, or that its structural integrity will be maintained, then a single train aligned in the SPC mode should be declared inoperable and its use in that mode restricted by the completion time specified for the applicable Limiting Condition for Operation (LCO) in the plant’s Technical Specifications.

Given that such analyses are required within the design basis, the staff has determined that the S&L analysis of the RHR system at LaSalle, as reported in EMD-067982, contains many simplifying assumptions for which the staff has identified numerous concerns that reflect on the adequacy of the waterhammer evaluation. The staff cannot verify that the RHR system will remain operable and/or functional following a LOOP/LOCA during operation in the SPC mode. It is the responsibility of the licensee to demonstrate the operability of the RHR system, and that its structural integrity will be maintained.

The staff recommends that the licensee address the specific findings in the above staff assessment to establish the RHR system operability as discussed in the provisions of Part 9900 of the NRC Inspection Manual. The staff also recommends that the licensee perform an evaluation of the dynamic loading and a detailed non-linear dynamic analysis of the RHR system, subject to ASME Code Section III, Appendix F, criteria. Conversely, the licensee should consider system modifications as a means of avoiding potential line voiding and subsequent waterhammer effects.

5.0 REFERENCES

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