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U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
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LaSalle County Station, Units 1 and 2
Facility Operating License Nos. NPF-11 and NPF-18
NRC Docket Nos. 50-373 and 50-374

Subject: Response to Task Interface Agreement 2001-14, "Evaluation of LaSalle Water Hammer Analysis"

References:

(1) Memorandum from G. E. Grant (NRC) to L. B. Marsh (NRC), "Task Interface Agreement (TIA 2001-14) Evaluation of LaSalle Waterhammer Analysis," dated November 2, 2001

(2) Memorandum from L. E. Marsh (NRC) to G. E. Grant (NRC), "Task Interface Agreement (2001-14) Evaluation of LaSalle Waterhammer Analysis (TAC NOS. MB3366 and MB3367)," dated July 1, 2002

The purpose of this letter is to provide Exelon Generation Company, LLC, (EGC) perspectives regarding a recently received Task Interface Agreement (TIA) 2001-14, "Evaluation of LaSalle Water Hammer Analysis," involving the impact of operating the LaSalle County Station (LSCS) Residual Heat Removal (RHR) system in Suppression Pool Cooling (SPC) mode. Notwithstanding the differing conclusions regarding this matter, EGC firmly believes that the previous operation of the RHR system in SPC mode does not represent a significant safety issue.

As background, during the Spring of 2001, LSCS experienced increased leakage from several Safety Relief Valves (SRVs). The discharge from the SRVs is directed to the primary containment suppression pool, which results in a heating of the suppression pool water. During plant operation, the RHR is normally aligned for Low Pressure Coolant Injection (LPCI); however, the RHR can be realigned for SPC.

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In May 2001, LSCS prepared an assessment to address the operation of the RHR in SPC mode for the summer of 2001. The assessment concluded that continuous operation of one train of RHR in the SPC mode was within the design basis for the station. We acknowledge that this assessment did not provide an adequate basis to support the planned operation in that it did not meet ASME code requirements. Based on the conclusions of this inadequate assessment, LSCS operated one train of RHR in SPC on Unit 1 almost continuously from May 25 to September 3, 2001.

During the Summer of 2001, the NRC expressed concerns about the extended use of the SPC mode. With Reference 1, the NRC Region III office requested the NRC Office of Nuclear Reactor Regulation (NRR) to review this use of the RHR at LSCS. The Reference 2 Memorandum provided the results of NRR's review in a TIA.

After reviewing the TIA response, LSCS initiated an interim action requiring that whenever a single train of RHR is operating in the SPC mode, the LPCI mode of operation for that RHR train is declared inoperable and Technical Specification 3.5.1, "ECCS – Operating," Condition and Required Action A.1 is entered. This requires restoration of the LPCI subsystem to operable status in 7 days.

EGC has subsequently completed a comprehensive review of the referenced document and concludes the following.

- EGC has determined that LSCS inappropriately credited a water hammer analysis that did not meet ASME Code requirements.
- EGC believes that the TIA position that a water hammer analysis is required regarding operation of RHR in SPC at LSCS, is inconsistent with the original design basis accepted by the NRC. Additionally, the TIA position is inconsistent with the results of previous staff reviews of water hammer potential (i.e., NRC NUREG-0927, "Evaluation of Water Hammer Occurrence in Nuclear Power Plants," Revision 1, dated March 1984 and AEOD Engineering Evaluation Report AEOD/E91-01, "A Review of Water Hammer Events after 1985," dated February 1991). The NRC did not require this water hammer analysis to demonstrate General Design Criteria (GDC) compliance during and since the initial licensing of LSCS. To do so at this time would constitute the imposition of a regulatory staff position interpreting the Commission rules with a new and different position from the previously applicable staff position. As such, LSCS requests that a backfit analysis be performed in accordance with 10 CFR 50.109, "Backfitting." The backfit analysis is required to demonstrate that there is a substantial increase in the overall protection of the public health and safety from the backfit, and that the direct and indirect costs of implementation for LSCS are justified in view of this increased protection.
- Finally, EGC concurs that the NRC's continuing concerns over the potential for a water hammer while in SPC mode is generic in nature, and should be resolved using existing generic resolution processes.

The details of our review of are provided in Attachment 1.

Due to the generic nature of this issue and the newly-articulated staff position that does not appear consistent with the initial design approval, EGC requests that this TIA position not be imposed at LSCS. EGC further requests that this issue be resolved generically and uniformly addressed within the industry.

Should you have any questions concerning this submittal, please contact Mr. T. W. Simpkin at (630) 657-2821.

Respectfully,

Keith R. Jury

Keith R. Jury
Director – Licensing
Mid-West Regional Operating Group

Attachments:

Attachment 1. Comments on TIA 2001-14

cc: Regional Administrator – NRC Region III
NRC Senior Resident Inspector – LaSalle County Station
Office of Nuclear Facility Safety – Illinois Department of Nuclear Safety
Deputy Director, Division of Systems Analysis and Regulatory Effectiveness – Office of Nuclear Regulatory Research

ATTACHMENT 1
Comments on TIA 2001-14

Overview and Background

During the Spring of 2001, LaSalle County Station (LSCS) Unit 1 experienced increased leakage from several Safety Relief Valves (SRVs). The increased leakage was within Technical Specifications (TS) leakage rates. The discharge from SRV leakage is directed to the primary containment suppression pool, which results in a heating of the suppression pool water. During plant operation, the Residual Heat Removal System (RHR) is normally aligned for Low Pressure Coolant Injection (LPCI). However, the RHR can be realigned for Suppression Pool Cooling (SPC).

In May 2001, LSCS Engineering prepared an assessment to address the operation of the RHR in SPC mode for an extended period of time. The assessment used a pre-existing water hammer evaluation and concluded that continuous operation of one train of RHR in SPC mode during the Summer of 2001 was within the plant's design basis. Based on this assessment, LSCS Unit 1 operated one train of RHR in SPC almost continuously from May 25 to September 3, 2001.

An NRC Inspector expressed concerns about this method of operation in August 2001. Multiple discussions were held between the NRC Inspector and LSCS personnel. The key issues involved the lack of a rigorous water hammer analysis of RHR in SPC mode, the time limit that RHR can be in SPC mode, and the operability of the LPCI mode of RHR during plant operation while in SPC mode. LSCS re-evaluated this issue, confirmed the original conclusion, and shared this conclusion with the NRC Inspector. On November 2, 2001, the NRC Inspector requested a NRC Task Interface Agreement (TIA) 2001-14, "Evaluation of LaSalle Water Hammer Analysis," be prepared to evaluate this issue (Reference 1).

Region III requested assistance from the NRC Office of Nuclear Reactor Regulation (NRR) to address two major issues. The first was a request for NRR to determine whether the continuous long-term operation of a single train of RHR in SPC mode was within the LSCS design basis. The second issue was to review the LSCS water hammer analysis to verify that the RHR system would remain operable and/or functional following a Loss of Offsite Power (LOOP) concurrent with a Loss of Coolant Accident (LOCA) during operation of the RHR in SPC mode.

On July 1, 2002, NRR responded to TIA 2001-14 (Reference 2). Key NRR conclusions in the TIA response and Exelon Generation Company, LLC, (EGC) perspectives on the NRC conclusions are provided below.

Continuous Long Term Operation of a Single Train of RHR in SPC Mode

The response to TIA 2001-14 states:

> "NRR has concluded that continuous long term operation of a single train of RHR system in the SPC mode is within the LaSalle design basis."

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Comments on TIA 2001-14

EGC Comments

EGC agrees with this conclusion. Currently, the LSCS licensing basis is silent with respect to any limitations on the use of SPC. However, the change that resulted in extended use of SPC did not meet the original design intent (i.e., that the RHR system operation in the SPC mode would constitute a small percentage of the total time), and the evaluation performed did not identify this condition. We have determined that the LSCS licensing basis needs to be clarified to prevent similar misunderstandings from occurring in the future. These clarifications are focused on the need and bases for SPC operation.

RHR SPC Water Hammer Analysis

The response to TIA 2001-14 states:

(1) "...the staff has determined that the LaSalle water hammer analysis contains many simplifying assumptions for which the staff has identified numerous concerns that reflect on the adequacy of the water hammer 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.",

(2) "If a licensee's analysis for water hammer 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.", and

(3) " RHR system analyses which demonstrate that the plant safety system can withstand a water hammer 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 analysis ensure that a consequential failure of LOOP/LOCA does not result in a loss of the capability of the RHR to perform its safety function."

EGC Comments to Statement 1

Statement 1:

(1) "...the staff has determined that the LaSalle water hammer analysis contains many simplifying assumptions for which the staff has identified numerous concerns that reflect on the adequacy of the water hammer 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."

ATTACHMENT 1
Comments on TIA 2001-14

In May 2001, LSCS prepared an assessment to address the operation of the RHR in SPC mode for the summer of 2001. The assessment used a water hammer analysis to support its conclusion that continuous operation of one train of RHR in the SPC mode was within the design basis for the station. Based on this assessment, LSCS operated one train of RHR in SPC on Unit 1 almost continuously from May 25, 2001 to September 3, 2001.

EGC agrees with the NRR position that the water hammer analysis used in this assessment did not provide adequate design basis justification for changing the original design intent regarding the extent of SPC usage to allow continuous operation of the RHR system in the SPC mode. While LSCS did not provide adequate justification for the intended operation, actual industry experience involving water hammers (discussed below) does support the overall conclusion of the analysis that, in the event of a LOOP/LOCA while in the SPC mode, functional capability will be retained.

In response to NRC Information Notice 87-10, the Boiling Water Reactor Owners' Group (BWROG) requested General Electric (GE) to investigate water hammer methodologies and the consequences of water hammer events that have occurred in nuclear power plants. The BWROG concluded in BWR Owners Group Report NEDC-32513, "Final Report on Suppression Pool Cooling and Water Hammer," dated December 29, 1995, that predictions of piping system response to water hammer loads tended to be unrealistically conservative because of conservatism in the modeling and assumptions. Actual water hammer events resulted in less severe damage than predictions and usually were limited to pipe hangers and mounts. The BWROG also noted that improvements to piping supports and mounts, coupled with initiatives by both the industry and regulatory agency, resulted in both a reduction of occurrence and consequences of water hammer events. This experience is further delineated below.

> The NRC reported in AEOD Engineering Evaluation Report AEOD/E91-01, "A Review of Water Hammer Events after 1985," dated February 1991, that 148 water hammer events had been reported from 1969 to 1980, another 40 events from 1981 to 1985 and 12 events thereafter up to March 1990. Of these 200 events, there were 16 Boiling Water Reactor (BWR) water hammer events involving flow into a voided RHR line. The NRC concluded in the report that none of these events resulted in damage to the reactor pressure boundary, loss of containment integrity, release of radioactivity outside the plant, or the inability of RHR to perform its intended safety function. The same conclusion had been reached earlier in Revision 1 of NRC NUREG-0927, "Evaluation of Water Hammer Occurrence in Nuclear Power Plants," dated March 1984. NUREG-0927, Revision 1 concluded that since 1969 (1969 to 1982), approximately 150 waterhammer events have been reported through the NRC Licensee Event Report process. Damage has been principally limited to pipe support systems. Only 12 events involved inoperability of safety-related components and all 12 events were due to flooding caused by waterhammer in non-safety related systems.
>
> Electric Power Research Institute (EPRI) report NP-6766, "Water Hammer Prevention, Mitigation, and Accommodation," July 1992 documents a total of 283 water hammer events in nuclear power plants from 1969 to May 1988, of which 160 occurred in BWRs. A total of 36 of these 160 events were associated with

> RHR systems. Of the 36 RHR related events, 12 were due to flow into a voided line. Damage caused by these events was generally limited to hangers, snubbers, and other supports. Only one event, a water hammer in 1974, resulted in a crack in the head spray line. The RHR system, including the head spray line, was determined to be operable and able to perform its intended function.

BWROG report NEDC-32513 noted that only one of the many water hammer events (i.e., April 1977) resulted in snubber damage from operation in the SPC mode. The BWROG concluded based on the historical and practical experience noted above, that none of the water hammer events posed a threat to public safety. Given the low probability of the postulated LOOP/LOCA with one train of RHR in the SPC mode sequence and the low likelihood that the water hammer would totally incapacitate the RHR system, it is concluded that a significant public risk does not exist. Therefore, the BWROG concluded that substantial additional effort to reduce the water hammer potential was not supported.

The subject of water hammer was originally identified as NRC Unresolved Safety Issue (USI) A-1, "Evaluation of Water Hammer Occurrences in Nuclear Power Plants." USI A-1 was considered resolved when Revision 1 of NUREG-0927 was issued in 1984. The reasons for closure of the USI item included the following.

> The frequency and severity of water hammer occurrences had been significantly reduced through design features such as keep-fill systems, vacuum breakers, J-tubes, void detection systems, improved venting procedures, increased operator awareness and training.

The NRC essentially confirmed this conclusion in 1991 in AEOD/E91-01. The NRC staff concluded: "...the frequency of water hammer events had decreased significantly since the initial review and that there were no new phenomena as causes of water hammer...new or additional requirements to reduce the number of water hammer events were not supported by cost-benefit guidelines."

Of the 26 events occurring in the RHR system of BWRs similar to that installed at LSCS addressed in NUREG-0927, 7 resulted in no damage, 18 resulted in support/ snubber damage only, and 1 resulted in a cracked head spray line at Hatch Nuclear Plant Unit 1 in 1974. Of the 13 events in the RHR systems of BWRs that resulted from the column-separation/ line-voiding water hammer mechanism, none resulted in pressure boundary leakage, and only one resulted in damage to a plant component.

Additional information is provided in EPRI TR-106438, "Water Hammer Handbook for Nuclear Plant Engineers and Operators," May 1996. This report documents 283 water hammers in U. S. plants between 1969 and 1998, of which 160 (i.e., approximately 56%) occurred in BWRs. Of the 12 events in the RHR systems of BWRs that resulted from the column-separation/line-voiding water hammer mechanism, none resulted in pressure boundary leakage, and only one resulted in damage to a plant component.

LSCS 1 has experienced two water hammer events in the RHR system.

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Comments on TIA 2001-14

- The first event occurred in September of 1982 and was documented in plant Deviation Report DVR-1-1-82-239. Prior to this event, the RHR system was being shutdown from the steam condensing mode of RHR and preparations were being made to fill and vent the RHR piping and heat exchangers. When a valve was opened to allow water into the "B" RHR heat exchanger, a loud water hammer was heard by plant personnel. After the water hammer had occurred, the filling and venting of the heat exchangers was completed and they were used for the suppression pool cooling mode of RHR. After suppression pool cooling was terminated, it was observed that a previously witnessed suppression pool level drop had stabilized. This fact, plus a loud water hammer experienced on "A" RHR heat exchanger the previous shift, led the shift personnel to believe that this heat exchanger had tube damage. A subsequent investigation concluded that heat exchanger integrity was maintained and no leakage occurred.

- The second water hammer in the LSCS RHR system occurred in August 1983 and was documented in plant Deviation Report DVR-1-1-83-403. This event was attributed to the presence of a steam bubble in loop "A" of the LPCI system when loop "A" RHR shutdown cooling was secured and partially depressurized when hot. The water hammer occurred when loop "A" RHR shutdown cooling was subsequently restarted. In October of 1983, a strut and eight snubbers were found damaged, and two sets of concrete expansion bolts as well as assorted pipe clamps were found loosened as a result of this water hammer. The point at which the damaged restraints were found corresponded to the high point in the loop "A" RHR LPCI system injection piping and aligned with the direction in which water hammer forces would be expected. This event is also documented in EPRI NP-6766.

  This second event is very similar to the LOOP/LOCA scenario postulated for LSCS, and is probably more severe since this event involved a steam void collapse. When loop "A" RHR was aligned for shutdown cooling, a significant flow path existed between the upper RHR piping and the suppression pool through the heat exchanger vent valve. Flow was significant enough to noticeably increase suppression pool level. Suppression pool cooling was terminated and re-initiated approximately 8 ½ hours later. This time would have been sufficient to drain all of the water out of the upper RHR piping such that an equilibrium void may have formed in the upper end of the piping. While the resulting water hammer caused damage to several pipe supports, the piping itself was undamaged and no pressure boundary failure occurred.

Industry reviews and operating experience support the LSCS conclusion that, in the event of a water hammer, the affected piping would remain capable of fulfilling its function.

LSCS has entered this issue into the Corrective Action Program while this matter is under discussion between the NRC and LSCS. However, taking this action should not be interpreted to mean that EGC agrees with the NRR stated regulatory position that a water hammer analysis to the ASME Code Section III, Appendix F criteria is required to operate RHR in SPC mode. EGC comments on this issue are described below.

ATTACHMENT 1
Comments on TIA 2001-14

EGC Comments to Statement 2

Statement 2:

(2) "If a licensee's analysis for water hammer 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."

As previously stated, although the analysis did not singularly provide adequate demonstration of SPC operability, industry reviews and operating experience support the LSCS conclusion that, in the event of a water hammer, the affected piping would remain capable of fulfilling its function. However, as noted, upon receipt of and review of the NRR review, LSCS initiated a conservative interim action pending the outcome of the generic review of this issue. Specifically, whenever a train of RHR is operating in the SPC mode, the LPCI mode of operation for that single RHR train is declared inoperable and Technical Specification 3.5.1, "ECCS – Operating," Condition and Required Action A.1 is entered. This requires restoration of the LPCI subsystem to operable status in 7 days.

Similar to other actions already discussed, this interim action should not be interpreted by the NRC to imply that EGC agrees with the NRR stated regulatory position that a water hammer analysis to the ASME Code Section III, Appendix F, criteria is required to operate RHR in SPC mode. Additional comments on this issue are described in detail below.

EGC Comments to Statement 3

Statement 3:

(3) "RHR system analysis which demonstrate that the plant safety system can withstand a water hammer 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 analysis ensure that a consequential failure of LOOP/LOCA does not result in a loss of the capability of the RHR to perform its safety function."

The Safety Evaluation attached to the TIA refers to a 2% limit as defining a SPC operating limit. EGC does not agree with the TIA response that the use of the 2% limit from Standard Review Plan 3.6.1, "Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," Branch Technical Position (BTP) MEB 3-1, "Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment," provides an operating limit for RHR in SPC mode. Additionally, EGC does not agree with the TIA response that the operation of RHR in SPC mode adds to the hours that RHR is operated as a high energy fluid system and the conclusions drawn by the NRC from this characterization. EGC also does not agree that there is a requirement to perform a water hammer analysis in this circumstance to demonstrate compliance with GDC 4, 17, and 35.

Applicable Regulatory Documents

The licensing requirements used for LSCS are the General Design Criteria (GDC) specified in Title 10 of the Code of Federal Regulations, Part 50, Appendix A. The TIA Response addressed three specific criteria: GDC 4, GDC 17, and GDC 35.

GDC 4 addresses environmental and dynamic effects that a system must be designed to accommodate. GDC 4 contains an exemption criterion for 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.

The criteria to qualify for this exemption are contained in BTP MEB 3-1. For systems that are exposed to both high energy and moderate energy fluids, the system can be considered as a moderate energy system if it operates as a high energy fluid system less than 2% of the time that the system operates as a moderate energy fluid system. The pressure-temperature conditions of the BTP MEB 3-1 specific to high-energy fluid system classification are as follows

$$>200°F \text{ or } >275 \text{ psig}$$

During the SPC mode of RHR operation, the temperature of the suppression pool water is limited to 105°F by Technical Specifications. Normal operating pressure of the RHR system while in SPC is between 68 and 150 psig. Thus, the operating conditions of the RHR in SPC mode are as follows.

$$\leq 105°F \text{ and } \leq 150 \text{ psig}$$

The operation of RHR in SPC mode adds to the total hours that the RHR operates as a moderate energy fluid system. As such, if the 2% limit specified in BTP MEB 3-1 is used as the operating limit for RHR in SPC mode, it would not restrict the time that RHR is in SPC mode as that operation is a moderate energy fluid operation.

EGC review of the NRR response did not reveal any issues associated with GDC 17 or 35 when RHR is in the SPC mode. EGC concurs that the operation of RHR in SPC mode meets the requirements of GDC 17 and 35.

NUREG 0519, "Safety Evaluation Report related to the operation of LaSalle County Station," dated March 1981, Section 3.6, Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping, states the following.

> "LaSalle is designed to withstand the effects of postulated pipe breaks and leakage cracks, including pipe whip, jet impingement and reaction forces, and environmental effects. The means used to protect safety-related systems and components include physical separation, enclosure within suitably designed structures, pipe whip restraints, and equipment shields. Protection against pipe failure outside containment is in

accordance with A. Giambusso's letter (NRC) dated December 12, 1972, "General Information Required for Consideration of the Effects of a Piping System Break Outside Containment," which is referenced in Section 3.6.1 of the Standard Review Plan and is supplemented by a moderate energy line break analysis in accordance with the guidelines of Branch Technical Position ASB 3-1, "Protection Against Postulated Piping Failure in Fluid Systems Outside Containment." The applicant has also presented an analysis on the effect of the moderate energy line breaks outside containment on safety-related systems. The moderate energy systems are designed to meet the criteria set forth in Branch Technical Position ASB 3-1. We have evaluated the analysis and conclude that a postulated pipe crack in a moderate energy line will not cause loss of function on any safety-related system.

The applicant has presented its methodology in the Final Safety Analysis Report for determining the location, type, and effects of postulated pipe breaks in high energy piping systems and postulated pipe cracks in moderate energy piping systems. Using these postulated events, the applicant evaluated its design of systems, components, and structures necessary to safely shut the plant down and to mitigate the effects of these postulated piping failures. In addition, it was indicated that pipe whip restraints, jet impingement barriers, and other such devices will be used to mitigate the effects of these postulated failures.

We reviewed these criteria stated in the Final Safety Analysis Report and conclude that they provide for a spectrum of postulated pipe breaks and pipe cracks which includes the most likely locations for piping failures, and that the types of breaks and their effects are conservatively assumed. We find that the methods used to design the pipe whip restraints provide adequate assurance that they will function properly in the event of a postulated piping failure. We further conclude that the use of the applicant's proposed pipe failure criteria in designing the systems, components, and structures necessary to safely shut the plant down and to mitigate the consequences of these postulated piping failures provide reasonable assurance of their ability to perform their safety function following a failure in high or moderate energy piping systems.

Major high-energy systems that the applicant analyzed included the main steam, feedwater, reactor core isolation cooling, condensate booster, high pressure coolant injection and the extraction steam systems. Initially, the applicant did not provide adequate justification for not analyzing the residual heat removal system as a high energy fluid system outside containment. The applicant in Amendment 41, at our request, showed by analysis that the residual heat removal system operations as a high energy system less than 2 percent of the time, therefore, according to our guidelines, it should be considered a moderate energy piping system. We have reviewed the applicant's analysis and agree with the results. We also conclude that a postulated pipe break during the high-energy portion of operation will not prevent the safe shutdown of the plant and, therefore, is acceptable. Based on our review, we find that the

> applicant has adequately designed and protected areas and systems required for safe plant shutdown following postulated events, including the combination of pipe failure and single active failure. The plant design meets the criteria set forth in Sections 3.6.12 and 3.6.2 of the Standard Review Plan; Branch Technical Positions MEB 3-1 and ASB 3-1, in regards to the protection of safety-related systems and components from a postulated high energy line break and in regards to the protection of safety-related systems and components from a postulated moderate energy line failure; and also complies with the applicable portions of Criteria 2 and 4 of the General Design Criteria. We, therefore, conclude the plant design for the protection of safety-related equipment against dynamic effects associated with the postulated rupture of piping is acceptable."

The above information supports LSCS position that the design of the RHR system meets the requirements of GDCs 4, 17 and 35. The current RHR system design is reflective of the design at initial licensing of LSCS, except that the steam condensing mode is no longer utilized.

The NRC did not require this water hammer analysis to demonstrate GDC compliance during and since the initial licensing at LSCS. To do so at this time would constitute the imposition of a regulatory staff position interpreting the Commission rules with a new and different position from the previously applicable staff position. As such, LSCS requests that a backfit analysis be performed in accordance with 10 CFR 50.109, "Backfitting." The backfit analysis is required to demonstrate that there is a substantial increase in the overall protection of the public health and safety from the backfit, and that the direct and indirect costs of implementation for LSCS are justified in view of this increased protection.

<u>Generic Versus LSCS Specific Issue</u>

The TIA concluded that:

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

<u>EGC Comments</u>

EGC agrees with this NRR statement. Furthermore, the NRC should not reach a final conclusion regarding LSCS until this generic matter has been assessed. The following information further confirms that if the NRC maintains its position on this matter, it would represent both a generic and a plant-specific change in position. EGC conducted an informal survey of BWR 3/4s and 5/6s to determine how this issue is addressed. The following are the results of that survey. Note that there appears to be a wide range of approaches taken by the industry on this matter. Therefore, for the NRC to conclude that, at LSCS, a water hammer analysis must be provided to adequately demonstrate RHR operability while in the SPC mode does not have consistent precedent.

- Eighteen BWRs responded to the survey.

- Two BWRs have water hammer analyses and do not declare the ECCS function of RHR inoperable while operating in the SPC mode.
- Seven BWRs responded that the ECCS function of RHR is considered operable due to the extremely low probability of the event and none have a water hammer analysis.
- Three BWRs responded that the ECCS function of RHR is considered operable due to the establishment of a specific time limit for operating in the SPC mode. All three have time limits greater than two percent and none have a water hammer analysis.
- Six of the eighteen responding BWRs declare the ECCS function of RHR inoperable while in the SPC mode for varying reasons.
  - Two of these six facilities declare the ECCS function of RHR inoperable due to vulnerabilities associated with LPCI Loop Select logic.
  - Three of these six facilities declare the ECCS function of RHR inoperable due to ECCS response time issues.
  - One facility declares the ECCS function of RHR inoperable due to not having a water hammer analysis, similar to the NRC position stated in the TIA response.

In addition to the above, EGC has discussed these issues with the Potential Issue Resolution Task Force (PIRT) of the BWROG. As a result of the discussions, a BWROG Ad Hoc Committee meeting is currently being scheduled early in 2003 to further discuss/address these issues.

Summary

EGC agrees that LSCS inappropriately credited a water hammer analysis that did not meet ASME Code requirements. This resulted in operation of the system for a period of time greater than the original design intent of the system. However, as demonstrated through the application of the Significance Determination Process and available industry operating experience, this did not constitute a safety significant deficiency.

EGC also believes that the TIA position that a water hammer analysis is required is inconsistent with the original design accepted for RHR in SPC at LSCS. Additionally, the TIA position is inconsistent with the results of previous staff reviews of water hammer potential (i.e., NRC NUREG-0927 and AEOD/E91-01). The NRC did not require this water hammer analysis to demonstrate General Design Criteria (GDC) compliance during and since the initial licensing at LSCS. To do so at this time would constitute the imposition of a regulatory staff position interpreting the Commission rules with a new and different position from the previously applicable staff position. As such, EGC requests that a backfit analysis be performed in accordance with 10 CFR 50.109, "Backfitting." The backfit analysis is required to demonstrate that there is a substantial increase in the overall protection of the public health and safety from the backfit, and that the direct and indirect costs of implementation for LSCS are justified in view of this increased protection.

Finally, we believe the continuing concerns over water hammer potential while in SPC mode is generic in nature, and should be resolved using existing generic resolution processes. EGC is an active participant in the BWROG effort on this subject.