UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

October 21, 1997

MEMORANDUM TO: J. A. Grobe, Acting Director
Division of Reactor Safety, Region III

FROM: Robert A. Capra, Director
Project Directorate III-2
Division of Reactor Projects III/IV, NRR

SUBJECT: TASK INTERFACE AGREEMENT AITS-96-0389 - QUAD CITIES, UNITS 1 AND 2, REGARDING NEDC-32513 APPLICABILITY TO RHR SYSTEM WATER HAMMER POTENTIAL (TAC NOS. M97323 AND M97324)

By memorandum dated November 18, 1996, Region III requested the Office of Nuclear Reactor Regulation (NRR) to answer the following question: "Was a mechanistic consequence such as a water hammer as a result of a LOCA concurrent with a LOOP a licensing requirement and intended in the original design and in the licensing review process?" In this evaluation the staff has referenced, but did not review in its entirety, a General Electric report which was attached to your November 18, 1996, memorandum; NEDC-32513, "Suppression Pool Cooling and Water Hammer," dated December 29, 1995. This report was prepared by the Boiling Water Reactor Owners Group to address concerns raised in NRC Information Notice (IN) 87-10. The staff has also issued IN 87-10, Supplement 1, dated May 15, 1997. It should be noted that NEDC-32513 contains a discussion of plant licensing basis pertaining to General Design Criteria (GDC) 17, portions of which we disagree. This evaluation only references this report while addressing your specific question and should not be considered an endorsement of the report.

In accordance with GDC 35, the staff has concluded that licensees are required to address unavailability of either onsite or offsite power (whichever is more limiting) concurrent with a LOCA and the consequences of the event. If the loss of offsite power is more limiting, the licensee is required to consider the LOOP concurrent with the LOCA. Specifically, while not licensed to the current GDCs, Quad Cities did address draft GDC 41 by stating the facility emergency core cooling systems are designed for concurrent postulated design basis reactor primary system breaks with the loss of all offsite ac power. The specific mechanism for water hammer requires that the suppression pool cooling (SPC) mode of residual heat removal (RHR) be in use at the time of the LOCA. It states in IN 87-10, Supplement 1, that since the probability of a water hammer event increases as

CONTACT: G. Golub, SRXB/DSSA
(301) 415-1329; E-Mail GRG

R. Pulsifer, DRPW/PDIII-2
(301) 415-3016; E-Mail RMP3

A/1

~~ACC# 9711060168~~ CF

the amount of time the system is operated in the SPC mode increases, and the likelihood of damage to the system increases with the frequency of water hammer events, operating in the SPC mode more often than assumed in the UFSAR may be an unreviewed safety question as defined in 10 CFR 50.59(a)(2)(i). In addition, a significant increase in the amount of time the RHR system is operated may affect the amount and types of preventive maintenance and monitoring activities that are required to ensure that it is capable of performing its intended function. Consequently, if licensees determine that the frequency of use of the SPC mode of RHR is greater than that assumed in the UFSAR then LOCA occurrence during SPC mode should be postulated and the corresponding draindown/water hammer should be addressed.

Docket Nos. 50-254, 50-265

Attachment: Safety Evaluation

cc: J. Wiggins, RI
J. Jaudon, RII
A. Howell, RIV

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

LICENSING BASIS REQUIREMENTS REGARDING

WATER HAMMER AS A RESULT OF A LOOP/LOCA

COMMONWEALTH EDISON COMPANY AND

MIDAMERICAN ENERGY COMPANY

QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2

DOCKET NOS. 50-254 AND 50-265

1.0 BACKGROUND

By memorandum dated November 18, 1996 (Reference 1), Region III requested the Office of Nuclear Reactor Regulation (NRR) to answer the following question: "Was a mechanistic consequence such as a water hammer as a result of a LOCA concurrent with a LOOP a licensing requirement and intended in the original design and in the licensing review process?" In this evaluation the staff has referenced but did not review in its entirety a General Electric report which was attached to your November 18, 1996, memorandum; NEDC-32513, "Suppression Pool Cooling and Water Hammer," (Reference 2). The report was generated for the Boiling Water Reactor (BWR) Owners' Group to address concerns raised in NRC Information Notice 87-10, "Potential for Water Hammer During Restart of Residual Heat Removal (RHR) Pumps" (Reference 3). The staff has also issued IN 87-10, Supplement 1, dated May 15, 1997. The specific condition in question involves a design basis loss-of-coolant accident (LOCA) and corresponding loss of offsite power (LOOP) occurring while part or all of the RHR system was aligned to suppression pool cooling (SPC) mode. During the power loss and subsequent valve realignment, portions of the RHR system will void because of draindown to the suppression pool as a result of elevation difference. A water hammer may occur in those RHR loops that were in the suppression pool cooling mode when the RHR pumps restart after the diesel generators reenergize the buses.

2.0 EVALUATION

2.1 BWR LOCA Design Basis Requirements

Appendix A to 10 CFR Part 50, General Design Criterion (GDC) 35, "Emergency Core Cooling," states that suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capability shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric

ATTACHMENT

power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure. Therefore, individual licensees must examine unavailability of either onsite or offsite power at the time of the LOCA to determine which scenario is limiting for the LOCA event. If the offsite power unavailability is more limiting, then the licensee must consider a LOOP concurrent with the LOCA. Consequently, the analysis assumes the onsite power supply (e.g., diesel generators) is available, but not operating. Therefore, in assuming a LOOP, the licensee must also address the time required following the LOCA/LOOP for diesel startup, connection to emergency core cooling systems (ECCS), and time for ECCS to reach rated flow. This time varies from 25 to 40 seconds following the LOCA/LOOP.

Quad Cities was not formally licensed to the current GDC's because the GDC's were not adopted as regulatory requirements at the time Quad Cities was built. However, the Quad Cities Updated Final Safety Analysis Report (UFSAR), Section 3.1, contains the licensee's understanding of the proposed GDC Criteria. The proposed GDC criterion was in draft form at the time of licensing of the Quad Cities facility. The UFSAR states; "Based on the applicant's understanding of the intent of the proposed criteria, it was felt that the Quad Cities station fully satisfies the intent of the criteria."

Draft GDC 41 - "ESF Performance Capability" is most equivalent to the current GDC 35 and it states that, "Engineered safety features such as emergency core cooling and containment heat removal systems shall provide sufficient performance capability to accommodate partial loss of installed capacity and still fulfill the required safety functions. As a minimum, each engineered safety feature shall provide this required safety function assuming a failure of a single active component." The response by the licensee states that; "The emergency core cooling systems (ECCS) are designed so that at least two different ECCS of different phenomena are provided to prevent clad melt over the entire spectrum of postulated design basis reactor primary system breaks. Such capability is available concurrently with the loss of all offsite ac power. The ECCS individual systems themselves are designed to various levels of component redundancy such that no single active component failure in addition to the accident will negate the required emergency core cooling capability."

Specifically, the Quad Cities UFSAR, page 15.6-13, states, "although the loss of water level or the increase in drywell pressure resulting from a pipe break is sensed immediately and the ECCS is signaled to start, the actual injection of water by the low pressure systems does not occur for about 30 seconds. This time is required for the diesel generators to start and accept load, the reactor pressure to fall below the ECCS pump discharge pressure and the ECCS pumps to achieve full flow. Water is injected into the reactor through both the [low-pressure core injection] LPCI system and the core spray system." Therefore, the LOCA/concurrent LOOP is within the Quad Cities design basis.

NEDC-32513 also addresses mechanistic coupling of the LOCA and LOOP. Mechanistic coupling of the two events considers the relative timing between

the LOCA event and the LOOP event. For the specific mechanism of water hammer under consideration, the timing between the LOCA and the LOOP would affect the magnitude of draindown if operating in SPC mode at the time of the LOCA. NEDC-32513 states that for the LOOP followed by a LOCA, draindown would occur between the time of the LOOP and the time that procedures require the operator to close the SPC return valve, approximately 30 minutes. Therefore, 30 minutes would represent the maximum time available for vessel draindown which is a precursor to water hammer in the RHR lines. NEDC-32513 states that mechanistic coupling of the two events is beyond the design basis. The staff has previously accepted LOCA analyses which assume a concurrent LOOP and licensees must continue to evaluate the concurrent LOCA/LOOP. Therefore, if SPC operation is more frequent than that assumed in the UFSAR, licensees may need to evaluate a water hammer event occurring due to draindown during the 25-40 seconds required for alignment of onsite power.

NEDC-32513, page 13, states, "it should be noted that for some plants it may be normal following a LOCA (without LOOP) to sequence the RHR pump (in SPC mode) onto an emergency bus. The delay in such a transfer could create conditions which may set up the potential for a water hammer. If this is the case, the water hammer potential remains the same while in the SPC mode during normal power operation if a LOCA occurs, whether there is a coincident LOOP or not." Therefore, if LOOP is not postulated on LOCA, individual licensees may need to consider draindown leading to water hammer, if LOCA occurrence during SPC mode is credible.

2.2 Frequency of use of RHR Suppression Pool Cooling Mode

The SPC mode of RHR is expected to be used during normal power operation to cool the suppression pool. Although operation in SPC mode was expected to be an infrequent occurrence, safety/relief valve (SRV) leakage has led to increased usage of SPC mode. Individual licensees should determine whether SPC is a significant mode of operation. Guidance is provided in IN 87-10, Supplement 1, on water hammer potential due to increased use of SPC.

Normally, in RHR standby mode, RHR pump discharge piping is kept in a filled condition to avoid water hammer and minimize time delay in LPCI injection when the RHR system is automatically initiated. Individual licensees should determine plant design bases and SPC operation to determine whether a LOCA during secondary modes of operation such as the SPC or test modes is credible, and whether the design basis has changed due to increased usage of SPC mode. In general, although the original design basis may not have considered a LOCA while in SPC mode, licensees may be required to do so based on frequency of use of SPC mode.

2.3 Safety Significance of RHR Water Hammer

The staff has previously reported on water hammer events and the safety significance of water hammer (References 3, 4, 5, 6). Specifically, NRC IN 87-10, "Potential for Water Hammer During Restart of Residual Heat Removal (RHR) Pumps," and IN 87-10, Supplement 1, states that other plants may have

high usage factors for SPC mode and large elevation differences in the RHR system, making those plants potentially subject to water hammer in the RHR system. NEDC-32513 discusses general conservatisms used in analysis of water hammer, but does not address the potential for or extent of draindown and the resulting water hammer for a concurrent LOCA/LOOP, a design basis event. If operation in SPC mode is beyond that assumed in the UFSAR then the licensee should address whether the magnitude of draindown following a coincident LOCA/LOOP could lead to water hammer in the RHR lines. If it is determined that a water hammer may exist, then the RHR lines should be analyzed to evaluate whether they can withstand such an event.

3.0 CONCLUSION

Individual licensees are required, in accordance with GDC 35, to address a concurrent LOCA/LOOP, if the unavailability of offsite power is more limiting than onsite power. Quad Cities in particular states in the UFSAR that they fully satisfy the intent of the proposed GDC criteria during the time of plant licensing. The licensee's response to draft GDC 41 states the ECCS is designed for the entire spectrum of postulated design basis reactor primary system breaks concurrent with LOOP. For the specific mode of water hammer under consideration, a LOCA/LOOP occurs during operation in SPC mode. Although LOCA occurrence during secondary modes of operation (such as SPC mode) may not have been included in the original design basis, increased use of SPC mode, beyond that assumed in the UFSAR would require analysis of the event and the corresponding draindown and water hammer.

Principal Contributor: G. Golub
R. Pulsifer

Date:

4.0 REFERENCES

1. Letter from Grant, G. E., to Roe, J. W., "Request for Technical Assistance: Quad Cities Licensing Basis Requirements Regarding Water Hammer as a Result of a LOOP/LOCA (AITS 96-0389), dated November 18, 1996.

2. NEDC-32513, "Suppression Pool Cooling and Water Hammer", dated December 1995.

3. NRC Information Notice 87-10, "Potential for Water Hammer During Restart of Residual Heat Removal (RHR) Pumps," dated February 11, 1987.

4. NRC Information Notice 91-50, "A Review of Water Hammer Events After 1985," dated August 20, 1991.

5. NRC Information Notice 85-76, "Recent Water Hammer Events," dated September 19, 1985.

6. NUREG-0927, Revision 1, "Evaluation of Water Hammer Occurrence in Nuclear Power Plants, Technical Findings Relevant to USI A-1," dated March 1984.