Entergy

Entergy Nuclear Northeast
Entergy Nuclear Operations, Inc.
Indian Point Energy Center
295 Broadway, Suite 1
P.O. Box 249
Buchanan, NY 10511-0249

January 31, 2003
NL-03-023

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop 0-P1-17
Washington, DC 20555

Subject: Indian Point 2 and 3 Nuclear Power Plants
Docket Nos. 50-247 and 50-286
Supplemental Response to Request For Additional Information Regarding NRC Generic Letter 96-06: Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions (TAC Nos M96822 and M96823)

References:

1. NRC Letter to Entergy Nuclear Operations, Inc; "Electric Power Research Institute (EPRI) Report TR-113594, "Resolution of Generic Letter 96-06 Waterhammer Issues," Volumes 1 and 2 -- Indian Point Nuclear Generating Unit Nos 2 and 3 (TAC Nos M96822 and M96823)" dated April 29, 2002.

2. NRC Generic Letter 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions," dated September 30, 1996 and supplement dated November 13, 1997.

3. Entergy letter to NRC, NL-02-105 / IPN-02-063; "Response to Request for Additional Information Regarding NRC Generic Letter 96-06: Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions" dated July 30, 2002

Dear Sir:

This letter provides a supplemental response to the NRC request for additional information (Reference 1) regarding the resolution of Generic Letter 96-06 (Reference 2) water hammer issues. The letter supplements the response previously provided by Entergy Nuclear Operations, Inc (Reference 3) for Indian Point Units 2 and 3.

The request for additional information, contained in Section 3.3 of the NRC SER for EPRI Report TR-113594, requested a supplement to previous responses to Generic Letter 96-06 for Indian Point Units 2 and 3 in three areas. Reference 3 provided information addressing two of the three areas. This letter provides information addressing the remaining area, which requests: "Certification that the EPRI methodology, including clarifications, was properly applied and that plant-specific risk considerations are consistent with the risk perspective that was

A072

provided in the EPRI letter of February 1, 2002." Entergy Nuclear Operations, Inc. certifies that the EPRI methodology, including clarifications, was properly applied and that plant-specific risk considerations are consistent with the risk perspective that was provided in the EPRI letter of February 1, 2002.

The NRC also requested, in Section 3.3 of the SER, that additional certification be provided for the probability of pipe failure if the uncushioned velocity at closure was more than 40 percent greater than the cushioned velocity. In-plant tests that simulated LOOP-only events were performed at both units. The test results demonstrated the integrity of the systems. The in-plant test data was also used for analytical qualification of the piping systems. In the in-plant LOOP-only tests, there was minimal waterhammer cushioning at the time of void closure because there was very little gas or steam in the void. Since this data was used for qualification for the LOCA or MSLB event the analysis for qualification for the LOCA or MSLB event took essentially no cushioning into account. Cushioning that would be expected due to accumulation of gas and steam in the void after a LOOP with a LOCA or MSLB event, would reduce the magnitude of the expected waterhammer. Since the analysis was performed using uncushioned velocity, the pipe failure probabilities provided in the aforementioned EPRI report remain bounding and the additional certification is not required.

Additional details are provided in Enclosures 1 and 2 for Indian Point Units 2 and 3, respectively. There are no new commitments made in this letter. If you have any questions, please contact Mr. Kevin Kingsley at (914) 734-5581.

I declare under penalty of perjury that the foregoing is true and correct. Executed on 1/31/03

Very truly yours,

Fred Dacimo
Site Vice President
Indian Point Energy Center

cc: next page

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Mr. Patrick D. Milano, Senior Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Mail Stop 0-8-C2
Washington, DC 20555-0001

U.S. Nuclear Regulatory Commission
Senior Resident Inspector's Office
Indian Point Unit 2
P.O. Box 38
Buchanan, NY 10511

U.S. Nuclear Regulatory Commission
Senior Resident Inspector's Office
Indian Point Unit 3
P.O. Box 337
Buchanan, NY 10511

Mr. Paul Eddy
NYS Department of Public Service
3 Empire Plaza
Albany, NY 12223

ENCLOSURE 1 TO NL-03-023

ADDITIONAL INFORMATION SUPPORTING SUPPLEMENTAL RESPONSE TO GENERIC LETTER 96-06 FOR INDIAN POINT 2

ENTERGY NUCLEAR OPERATIONS, INC.
INDIAN POINT NUCLEAR GENERATING UNITS NO. 2 AND 3
DOCKET NO. 50-247 AND 50-286

Summary

Through in-plant testing and structural analysis, the Indian Point Unit 2 Power Plant has been evaluated for the issues from Generic Letter 96-06 [6][1]. This letter certifies that the EPRI methodology, including clarifications, was properly applied, and that plant-specific considerations are consistent with the risk perspective that was provided in the EPRI letter dated February 1, 2002 [9, also included in Ref. 3]. The work for IP2 was reported in References 1, 2, and 7. The in-plant tests and analyses that were performed have been compared to the methods endorsed by the NRC in a Safety Evaluation Report [5] that accepted methods of evaluation developed by EPRI [3 and 4]. This report demonstrates that the methodology used in the tests and analysis at IP2 was consistent with the EPRI guidelines [3] for analyzing column closure waterhammer (CCWH) and condensation induced waterhammer (CIWH) at IP2.

Comprehensive tests [1] were performed in the plant for a loss of offsite power (LOOP) event. The system integrity was fully acceptable following these tests. This confirmed the capability of the IP2 service water system to withstand the CCWH that would result following the restart of the pumps. It was shown in the EPRI report and Reference 7 that the LOOP alone would be more severe than the LOOP occurring simultaneously with a LOCA or MSLB. The EPRI report that was accepted by the NRC stated that an acceptable method of qualification was to show that the system was acceptable by in-plant testing. This qualification by test methodology was used to show that the IP2 system performed acceptably.

The results of these in-plant tests were further used in an analysis of the piping and support system. The in-plant test results were used to characterize the pressure pulse that was then used in the analysis. The analysis showed that the piping, components, and supports in the plant were acceptable and met the applicable stress limits.

The service water system at IP2 was shown both by in-plant testing and by structural analysis to be acceptable. The methods used to verify system acceptability has been evaluated and shown to be consistent with guidance provided in the EPRI reports [3, 4].

The possible occurrence of a simultaneous LOCA/LOOP event has been evaluated and shown to be consistent with the risk perspective that was provided in the EPRI letter [9].

[1] Numbers in [] refer to the references provided at the end of the report.

Background

The service water (SW) system at Indian Point Unit 2 (PWR) is an open loop system that is served by a maximum of three pumps. There are five fan coolers at an elevation of approximately 68 feet above the river level. A vacuum at the coolers will only support 34 feet of water above the river. The coolers are supplied by and discharge through 10" lines that join in an 18" header with throttled butterfly valves prior to entering a 24" header that discharges into the Hudson River (elevation zero at mean low river level). During a LOOP, the pumps and fans lose power and the backup diesel generators start. The SW pumps restart approximately 25 seconds after initiation of a LOOP [7].

References 1, 2, and 7 are technical reports that address the effects of a Loss of Offsite Power (LOOP) event alone and concurrent with a Loss of Coolant Accident (LOCA) or Steam Line Break (SLB) on the Indian Point Unit 2 service water system. Reference 1 provides measured piping pressures, support loads, flows, and accelerations, based on monitoring of the service water (SW) fan cooler unit (FCU) supply and return lines during Safety Injection (SI) testing [8]. Reference 2 uses the data collected in Reference 1 to qualify the SW system for the issues from Generic Letter 96-06.

In-Plant Testing Program

The purpose of monitoring the SW system was to collect field data related to waterhammers in Containment Fan Cooler Unit (FCU) supply and return piping. As such, the SW system was monitored during a simulation of the "Safety Injection System, Electrical Load Test" [8]. This test simulates a LOOP. During LOOP conditions, a waterhammer is expected to occur due to column separation and column rejoining upon pump restart. The safety injection system test is performed during every refueling outage. A waterhammer would be expected to occur during each test.

Field data was obtained using pressure transducers, externally mounted strain gages, accelerometers, and flow meters. The measurements included pressure, flow, strain, and acceleration at selected locations in the system. Evaluation of the piping system [2] established that the SW piping to FCU No. 24 is representative of the other SW supply and return lines. Lines 11c and 12c that go to FCU No. 24 were determined to be the enveloping lines due to their longer lengths than the other sections of the piping. Longer pipe segments are more susceptible to pressure pulse loading than shorter pipe segments since the unbalanced loads from a passing pressure wave will be larger [Ref. 3, Section 6.4 "Structural Loading"]. The sensors were therefore located on SW Supply Line No. 11c and SW Return Line No. 12c to FCU No. 24. The test data was electronically recorded for subsequent analysis.

The piping system, including the pipe supports and equipment, was inspected and was acceptable following the test. Future ISI inspections will insure that piping supports are inspected on a periodic basis per ASME Section XI requirements.

Condensation Induced Waterhammer

Section 4.2 of the EPRI report [3] states that CIWH in low pressure service water systems is bounded by CCWH as long as the following limitations are met:

a) The system pressure at the time of the postulated CIWH is less than 20 psig.

   During the time that the pumps are off and a CIWH may occur, the maximum system pressure is less than 8 psia [7]. Therefore, this criterion is satisfied.

b) The piping system has been shown by test, analysis, or operating experience to be capable of withstanding a CCWH following LOOP, LOOP/LOCA, or LOOP/MSLB.

   The system has been tested numerous times and shown to withstand LOOP. Additionally, it has been shown analytically [2] that the system can withstand the simultaneous occurrence of a LOOP and a LOCA or MSLB.

Therefore, the criteria of the EPRI reports are satisfied and the CIWH is bounded by the CCWH at IP2.

Column Closure Waterhammer Analysis

The CCWH analysis of IP2 was based on characterization of the pressure pulses that were recorded during the simulated safety injection test [1]. The pressure pulse magnitude and pulse shape were developed from the test data. These pressure pulses were then used to develop a force time history. This force time history was then input to structural models of the piping system [2]. Fluid Structural Interaction (FSI) was not applied in this analysis. Air and steam cushioning that would be present with a LOCA or MSLB was not credited because the pressure pulse data was taken from a LOOP-only test. A LOOP-only event is essentially a non-cushioned event [9].

In the analysis, a sonic velocity of 2,300 ft/sec was used to propagate the pressure pulse through the pipe rather than the sonic velocity of 4,600 ft/sec recommended by the EPRI report. The use of a lower sonic velocity in the analysis only has the potential to affect the unbalanced piping forces because the pressure pulse magnitude and rise time was determined by the in-plant test data. The use of a lower sonic velocity results in equal or higher load magnitudes on an individual piping section, depending on the length of the piping section [Ref. 3, Section 6.4 "Structural Loading"]. In long piping sections, the magnitude of the load would be expected to be the same since the differential pressure will become fully developed independent of the wave speed. In short pipe sections, which are much more prevalent in the plant, the pipe load magnitude would be smaller with a higher sonic velocity because the front of the pressure wave reaches the other end of the pipe more quickly and, upon its arrival, will begin to reduce the differential pressure on the section. The use of a lower sonic velocity also will make the duration of the load longer on every pipe section. The longer duration of the pressure pulse results from the longer time that is required for the wave to pass through a pipe section at a slower wave speed. This effect on the unbalanced forces with a sonic velocity of 2,300 ft/sec will provide higher piping stresses and higher support forces than would be expected with a sonic velocity of

4,600 ft/sec. Therefore, the methodology used to apply the test data to the structural analysis of the IP2 SW system is appropriate to provide a bounding analysis.

The analysis [2] showed that the SW piping pipe supports, fan nozzles, containment penetrations, and fan cooler tubing were all acceptable.

The criteria of the EPRI reports are satisfied and the CCWH is shown to be acceptable by test data and analysis at IP2.

Risk Consideration

The User's Manual and the Technical Basis Report considered the risk of an unacceptable event occurring as a result of a LOOP/LOCA or LOOP/MSLB event. The conclusion was that the risk of an unacceptable event from the combination of the LOOP and LOCA or MSLB was small and that the methods in the User's Manual were suitable to demonstrate plant acceptability. The NRC concurred with this conclusion [5]. Assurance that the IP2 specific risk considerations are consistent with the NRC accepted risk perspective is provided below.

The EPRI report described the "progression" of events that could lead to an unacceptable condition. Since the SW system's safety function is to provide containment boundary, the "unacceptable condition" following a LOOP/LOCA or LOOP/MSLB event is defined as a breach of the pressure boundary. The events defined were as follows with a comparison to the IP2 conditions:

1. **Occurrence of a LOCA or MSLB** – NUREG/CR-5750 states that for a PWR, the mean frequency of occurrence of a large LOCA is $5x10^{-6}$/year and a medium LOCA is $4x10^{-5}$/year. The frequency of MSLB is $1x10^{-3}$/year. These generic values are considered appropriate for use in this evaluation with respect to the IP2 plant.

2. **Occurrence of a LOOP following a LOCA or MSLB** – Studies provided in NUREG/CR-6538 indicate that the dependent probability of a Loss of Offsite Power event following a LOCA event in a PWR is $1.4x10^{-2}$/demand. This value is considered applicable to the IP2 plant.

3. **Occurrence of a Simultaneous LOCA/LOOP or MSLB/LOOP Event** – The frequency of the combined event depends upon the probability of the LOCA or the MSLB and the dependent probability of the LOOP given that the LOCA or MSLB has occurred. Using the values defined in each of the NUREGs referenced above, the probability of the combined event is on the order of $1.5x10^{-5}$/year. The probability of the combined event referenced in the EPRI reports was also $1.5x10^{-5}$/year. Based on the same assumptions used in the EPRI report, the event combination is as likely at IP2 as that used in the evaluation of risk in the EPRI report and the SER.

4. **Void Formation** – The EPRI report concluded that, in an open loop plant, void formation would occur with the occurrence of a LOOP or a LOOP with a LOCA or MSLB if the elevation difference between the fan cooler and the supply or discharge piping is sufficient. At IP2, it is accepted that if a void forms, a waterhammer will occur.

5. **Pump Restart** – The EPRI report stated that the pumps will restart with certainty and the velocity of the fluid in the pipe, immediately prior to closing the void, will be defined by the pressure in the void, the piping geometry, and the pump characteristics. This uncushioned closure velocity can be reliably calculated. This velocity will not be higher than the rate at which the pumps, once restarted, can pump water. The calculation of the water velocity prior to closure is a plant specific analysis that can be conservatively performed. This is consistent with the situation at IP2.

6. **Column Closure** – The water columns will refill the void and the velocity at closure cannot be larger than the largest calculated differential velocity for the upstream and downstream water columns. This is consistent with the situation at IP2.

7. **Maximum Waterhammer Pressure** – The situation at IP2 is the same as that described in the EPRI report. Specifically, an upper bound on the waterhammer pressure can be calculated by the Joukowski relationship with the uncushioned closure velocity that corresponds to the pipe in which the closure will occur. The waterhammer pressure cannot be larger.

8. **Cushioned Waterhammer** – The IP2 waterhammer pressure is based on the measured CCWH pressure in a LOOP only test. Waterhammer following a LOOP-only event such as occurred in the in-plant test is not cushioned by gas or steam in a void. Using the IP2 waterhammer pressure, the piping stress code limits are not exceeded in the IP2 piping. Since essentially no cushioning is credited in the IP2 analysis, the probability of failure of the pipe is lower than calculated in the EPRI report. The probability of pipe rupture that was used in the EPRI reports, 10^{-2} per event, is considered to be a conservative estimate of the probability of pipe rupture for IP2.

9. **Likelihood of an Unacceptable Event** – Given the low frequency ($1.5x10^{-5}$/year) of the initiating events at IP2 and the low, but conservative, probability (10^{-2}) of piping failure, the use of the methodology in the User's Manual and the Technical Basis Report will lead to a likelihood of an unacceptable event that is on the order of $1.5x10^{-7}$. This probability is below the threshold for significant risk to the plant.

The specific risk of the events at IP2 is as likely as the risk provided in the EPRI reports. Hence, from the risk-informed perspective, the methods proposed in the EPRI Technical Basis Report and User's Manual are considered appropriate for use in this evaluation with respect to the IP2 plant.

References

1) Altran Technical Report No. 97143-TR-01, Revision 0, "Monitoring of A Service Water FCU Supply & Return Line During a System Electrical Load Safety Injection Test", prepared for Consolidated Edison Company of New York, Inc., Indian Point Unit 2, August 1997.

2) Altran Technical Report No. 97143-TR-02, Revision 1, "Design Basis Evaluation of Containment Fan Cooler Service Water Supply Line 11c and Return Line 12c", prepared for Consolidated Edison Company of New York, Inc., Indian Point Unit 2 Nuclear Power Station, September 1997.

3) EPRI Report 1006456, "Generic Letter 96-06 Waterhammer Issues Resolution User's Manual" - Proprietary, Final Report, April 2002.

4) EPRI Report 1003098, "Generic Letter 96-06 Waterhammer Issues Resolution, Technical Basis Report" - Proprietary, Final Report, April 2002.

5) Safety Evaluation Report, NRC Acceptance of the EPRI Report TR-113594 "Resolution of Generic Letter 96-06 Waterhammer Issues", Volumes 1 and 2, John Hannon to Vaughn Wagoner, April 3, 2002 (note that the number of the EPRI reports changed after issue of this SER).

6) USNRC Generic Letter 96-06, "Assurance of Equipment Operability and Containment Integrity during Design-Basis Accident Conditions", September 30, 1996.

7) Altran Technical Report 96233-TR-01, Revision 2, "Containment Fan Cooler Response to a Simultaneous LOCA & LOOP Event", prepared for Consolidated Edison of New York Indian Point Unit 2 Nuclear Power Station, January 2003.

8) Consolidated Edison Indian Point Station Test Analysis Performance Procedure PT-R14, "Safety Injection System Electrical Load Test", Rev. 14.

9) Letter dated February 1, 2002, from EPRI to the NRC Document Control Desk (Attention Mr. Jim Tatum), "Response to ACRS Comments (letter dated 10/23/01) on the EPRI Report on Resolution of NRC GL 96-06 Waterhammer Issues".

ENCLOSURE 2 TO NL-03-023

ADDITIONAL INFORMATION SUPPORTING SUPPLEMENTAL RESPONSE TO GENERIC LETTER 96-06 FOR INDIAN POINT 3

ENTERGY NUCLEAR OPERATIONS, INC.
INDIAN POINT NUCLEAR GENERATING UNITS NO. 2 AND 3
DOCKET NO. 50-247 AND 50-286

Summary

Through in-plant testing and structural analysis, the Indian Point Unit 3 Power Plant has been evaluated for the issues from Generic Letter 96-06 [8][2]. This letter certifies that the EPRI methodology, including clarifications, was properly applied, and that plant-specific considerations are consistent with the risk perspective that was provided in the EPRI letter dated February 1, 2002 [9, also included in Ref. 5]. The work for IP3 was reported in References 1, 2, 3, and 4. The in-plant tests and analyses that were performed have been compared to the methods endorsed by the NRC in a Safety Evaluation Report [7] that accepted methods of evaluation developed by EPRI [5 and 6]. This report demonstrates that the methodology used in the tests and analysis at IP3 was consistent with the EPRI guidelines [5] for analyzing column closure waterhammer (CCWH) and condensation induced waterhammer (CIWH) at IP3.

Comprehensive tests [10] were performed in the plant for a loss of offsite power (LOOP) event. The system integrity was fully acceptable following these tests. This confirmed the capability of the IP3 service water system to withstand the CCWH that would result following the restart of the pumps. It was shown in the EPRI report and Reference 4 that the LOOP alone would be more severe than the LOOP occurring simultaneously with a LOCA or MSLB. The EPRI report that was accepted by the NRC stated that an acceptable method of qualification was to show that the system was acceptable by in-plant testing. This qualification by test methodology was used to show that the IP3 system performed acceptably.

The results of these in-plant tests were further used in an analysis of the piping and support system. The in-plant test results were used to characterize the pressure pulse that was then used in the analysis. Even though the in-plant tests showed that the system performed acceptably, the analysis showed that ten supports required modification in order to assure that the design basis margins were maintained. Following these modifications, the piping, components, and supports in the plant were acceptable and met the applicable stress limits.

The service water system at IP3 was shown both by in-plant testing and by structural analysis to be acceptable. The methods used to verify system acceptability has been evaluated and shown to be consistent with guidance provided in the EPRI reports [5, 6].

The possible occurrence of a simultaneous LOCA/LOOP event has been evaluated and shown to be consistent with the risk perspective that was provided in the EPRI letter [9].

---

[2] Numbers in [] refer to the references provided at the end of the report.

Background

The service water (SW) system at Indian Point Unit 3 (PWR) is an open loop system that is served by a maximum of three pumps. There are five fan coolers at an elevation of approximately 68 feet above the river level. A vacuum at the coolers will only support 34 feet of water above the river. The coolers are supplied by and discharge through 10" lines that join in an 18" header with throttled butterfly valves prior to entering a 24" header that discharges into the Hudson River (elevation zero at mean low river level). During a LOOP, the pumps and fans lose power and the backup diesel generators start. The SW pumps restart approximately 25 seconds after initiation of a LOOP [11].

References 1, 2, 3, and 4 are technical reports that address the effects of a Loss of Offsite Power (LOOP) event alone and concurrent with a Loss of Coolant Accident (LOCA) or Steam Line Break (SLB) on the Indian Point Unit 3 service water system. Reference 1 provides measured piping pressures, support loads, flows, and accelerations, based on monitoring of the service water (SW) fan cooler unit (FCU) supply and return lines during Safety Injection (SI) testing [10]. References 2 and 3 use the data collected in Reference 1 to qualify the SW system for the issues from Generic Letter 96-06.

In-Plant Testing Program

The purpose of monitoring the SW system was to collect field data related to waterhammers in Containment Fan Cooler Unit (FCU) supply and return piping. As such, the SW system was monitored during the performance of the safety injection (SI) test [10]. This test simulates a LOOP. During LOOP conditions, a waterhammer is expected to occur due to column separation and column rejoining upon pump restart. The safety injection system test is performed during every refueling outage. A waterhammer would be expected to occur during each test.

Field data was obtained using pressure transducers, externally mounted strain gages, accelerometers, and flow meters. The measurements included pressure, flow, strain, and acceleration at selected locations in the system. Evaluation of the piping system [2 and 3] established that the SW piping to FCU No. 34 is representative of the other SW supply and return lines. Lines 11c and 12c that go to FCU No. 34 were determined to be the enveloping lines due to their longer lengths than the other sections of the piping. Longer pipe segments are more susceptible to pressure pulse loading than shorter pipe segments since the unbalanced loads from a passing pressure wave will be larger [Ref. 5, Section 6.4 "Structural Loading"]. The sensors were therefore located on SW Supply Line No. 11c and SW Return Line No. 12c to FCU No. 34. The test data was electronically recorded for subsequent analysis.

The piping system, including the pipe supports and equipment, was inspected and was acceptable following the test. Future ISI inspections will insure that piping supports are inspected on a periodic basis per ASME Section XI requirements.

Condensation Induced Waterhammer

Section 4.2 of the EPRI report [5] states that CIWH in low pressure service water systems is bounded by CCWH as long as the following limitations are met:

c) The system pressure at the time of the postulated CIWH is less than 20 psig.

   During the time that the pumps are off and a CIWH may occur, the maximum system pressure is 12 psia [4]. Therefore, this criterion is satisfied.

d) The piping system has been shown by test, analysis, or operating experience to be capable of withstanding a CCWH following LOOP, LOOP/LOCA, or LOOP/MSLB.

   The system has been tested numerous times and shown to withstand LOOP. Additionally, it has been shown analytically [2 and 3] that the system can withstand the simultaneous occurrence of a LOOP and a LOCA or MSLB.

Therefore, the criteria of the EPRI reports are satisfied and the CIWH is bounded by the CCWH at IP3.

Column Closure Waterhammer Analysis

The CCWH analysis of IP3 was based on a characterization of the pressure pulses that were recorded during the simulated safety injection test [1]. The pressure pulse magnitude and pulse shape were developed from the test data. These pressure pulses were then used to develop a force time history. This force time history was then input to structural models of the piping system [2 and 3]. Fluid Structural Interaction (FSI) was not applied in this analysis. Air and steam cushioning that would be present with a LOCA or MSLB was not credited because the pressure pulse data was taken from a LOOP-only test. A LOOP-only event is essentially a non-cushioned event [9].

In the analysis, a sonic velocity of 2,300 ft/sec was used to propagate the pressure pulse through the pipe, rather than the sonic velocity of 4,600 ft/sec recommended by the EPRI report. The use of a lower sonic velocity in the analysis only has the potential to affect the unbalanced piping forces because the pressure pulse magnitude and rise time was determined by the in-plant test data. The use of a lower sonic velocity results in equal or higher load magnitudes on an individual piping section, depending on the length of the piping section [Ref. 5, Section 6.4 "Structural Loading"]. In long piping sections, the magnitude of the load would be expected to be the same since the differential pressure will become fully developed independent of the wave speed. In short pipe sections, which are much more prevalent in the plant, the pipe load magnitude would be smaller with a higher sonic velocity because the front of the pressure wave reaches the other end of the pipe more quickly and, upon its arrival, will begin to reduce the differential pressure on the section. The use of a lower sonic velocity also will make the duration of the load longer on every pipe section. The longer duration of the pressure pulse results from the longer time that is required for the wave to pass through a pipe section at a slower wave speed. This effect on the unbalanced forces with a sonic velocity of 2,300 ft/sec will provide higher piping stresses and higher support forces than would be expected with a sonic velocity of 4,600 ft/sec. Therefore, the methodology used to apply the test data to the structural analysis of the IP3 SW system is appropriate to provide a bounding analysis.

The analysis [2 and 3] showed that ten supports required modification and these modifications were made. Following those modifications, the SW piping, pipe supports, fan nozzles, containment penetrations, and fan cooler tubing were all acceptable.

The criteria of the EPRI reports are satisfied and the CCWH is shown to be acceptable by test data and analysis at IP3.

Risk Consideration

The User's Manual and the Technical Basis Report considered the risk of an unacceptable event occurring as a result of a LOOP/LOCA or LOOP/MSLB event. The conclusion was that the risk of an unacceptable event from the combination of the LOOP and LOCA or MSLB was small and that the methods in the User's Manual were suitable to demonstrate plant acceptability. The NRC concurred with this conclusion [7]. Assurance that the IP3 specific risk considerations are consistent with the NRC accepted risk perspective is provided below.

The EPRI report described the "progression" of events that could lead to an unacceptable condition. Since the SW system's safety function is to provide containment boundary, the "unacceptable condition" following a LOOP/LOCA or LOOP/MSLB event is defined as a breach of the pressure boundary. The events defined were as follows with a comparison to the IP3 conditions:

1. **Occurrence of a LOCA or MSLB** – NUREG/CR-5750 states that for a PWR, the mean frequency of occurrence of a large LOCA is 5×10^{-6}/year and a medium LOCA is 4×10^{-5}/year. The frequency of MSLB is 1×10^{-3}/year. These generic values are considered appropriate for use in this evaluation with respect to the IP3 plant.
2. **Occurrence of a LOOP following a LOCA or MSLB** – Studies provided in NUREG/CR-6538 indicate that the dependent probability of a Loss of Offsite Power event following a LOCA event in a PWR is 1.4×10^{-2}/demand. This value is considered applicable to the IP3 plant.
3. **Occurrence of a Simultaneous LOCA/LOOP or MSLB/LOOP Event** – The frequency of the combined event depends upon the probability of the LOCA or the MSLB and the dependent probability of the LOOP given that the LOCA or MSLB has occurred. Using the values defined in each of the NUREGs referenced above, the probability of the combined event is on the order of 1.5×10^{-5}/year. The probability of the combined event referenced in the EPRI reports was also 1.5×10^{-5}/year. Based on the same assumptions used in the EPRI report, the event combination is as likely at IP3 as that used in the evaluation of risk in the EPRI report and the SER.
4. **Void Formation** – The EPRI report concluded that, in an open loop plant, void formation would occur with the occurrence of LOOP or a LOOP with a LOCA or MSLB if the

elevation difference between the fan cooler and the supply or discharge piping is sufficient. At IP3, it is accepted that if a void forms, a waterhammer will occur.

5. **Pump Restart** – The EPRI report stated that the pumps will restart with certainty and the velocity of the fluid in the pipe, immediately prior to closing the void, will be defined by the pressure in the void, the piping geometry, and the pump characteristics. This uncushioned closure velocity can be reliably calculated. This velocity will not be higher than the rate at which the pumps, once restarted, can pump water. The calculation of the water velocity prior to closure is a plant specific analysis that can be conservatively performed. This is consistent with the situation at IP3.
6. **Column Closure** - The water columns will refill the void and the velocity at closure cannot be larger than the largest calculated differential velocity for the upstream and downstream water columns. This is consistent with the situation at IP3.
7. **Maximum Waterhammer Pressure** – The situation at IP3 is the same as that described in the EPRI report. Specifically, an upper bound on the waterhammer pressure can be calculated by the Joukowski relationship with the uncushioned closure velocity that corresponds to the pipe in which the closure will occur. The waterhammer pressure cannot be larger.
8. **Cushioned Waterhammer** - The IP3 waterhammer pressure is based on the measured CCWH pressure in a LOOP only test. Waterhammer following a LOOP-only event such as occurred in the in-plant test is not cushioned by gas or steam in a void. Using the IP3 waterhammer pressure, the piping stress code limits are not exceeded in the IP3 piping. Since essentially no cushioning is credited in the IP3 analysis, the probability of failure of the pipe is lower than calculated in the EPRI report. The probability of pipe rupture that was used in the EPRI reports, 10^{-2} per event, is considered to be a conservative estimate of the probability of pipe rupture for IP3.
9. **Likelihood of an Unacceptable Event** - Given the low frequency (1.5×10^{-5}/year) of the initiating events at IP3 and the low, but conservative, probability (10^{-2}) of piping failure, the use of the methodology in the User's Manual and the Technical Basis Report will lead to a likelihood of an unacceptable event that is on the order of 1.5×10^{-7}. This probability is below the threshold for significant risk to the plant.

The specific risk of the events at IP3 is as likely as the risk provided in the EPRI reports. Hence, from the risk-informed perspective, the methods proposed in the EPRI User's Manual and the Technical Basis Report are considered appropriate for use in this evaluation with respect to the IP3 plant.

References

1. Altran Technical Report No. 97140-TR-01, Revision 0, "Monitoring of a Service Water FCU Supply & Return Line during a Simulated Safety Injection Test", prepared for New York Power Authority, Indian Point Unit 3, June 1997.
2. Altran Technical Report No. 97124-TR-01, Revision 3, "Structural Analysis of Containment Fan Cooler Supply and Return Lines Subjected to Waterhammer Loading", prepared for New York Power Authority, Indian Point Unit 3 Nuclear Power Station, June 1997.
3. Altran Technical Report No. 97124-TR-03, Revision 0, "Structural Evaluation of Supports on Containment Fan Cooler Supply and Return Lines Subjected to Waterhammer Loading", prepared for New York Power Authority, Indian Point Unit 3 Nuclear Power Station, June 1997.
4. Altran Technical Report No. 97108-TR-01 Revision 4, "IP-3 Service Water/Containment Fan Cooler Waterhammer Analysis", prepared for New York Power Authority, Indian Point Unit 3 Nuclear Power Station, July 1997.
5. EPRI Report 1006456, "Generic Letter 96-06 Waterhammer Issues Resolution, User's Manual", Proprietary, Final Report, April 2002.
6. EPRI Report 1003098, "Generic Letter 96-06 Waterhammer Issues Resolution, Technical Basis Report", Proprietary, Final Report, April 2002.
7. Safety Evaluation Report, NRC Acceptance of the EPRI Report TR-113594 "Resolution of Generic Letter 96-06 Waterhammer Issues", Volumes 1 and 2, John Hannon to Vaughn Wagoner, April 3, 2002 (note that the number of the EPRI reports changed after issue of this SER).
8. USNRC Generic Letter 96-06, "Assurance of Equipment Operability and Containment Integrity during Design-Basis Accident Conditions", September 30, 1996.
9. Letter dated February 1, 2002, from EPRI to the NRC Document Control Desk (Attention Mr. Jim Tatum), "Response to ACRS Comments (letter dated 10/23/01) on the EPRI Report on Resolution of NRC GL 96-06 Waterhammer Issues".
10. New York Power Authority IP3 Procedure No. 3PT-R003D, Rev. 2, "Safety Injection Test".
11. New York Power Authority IP3 Calc. No. IP3-CALC-ED-01131, "480V Interlock Setpoint Adequacy", Rev. 1.