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SUBJECT: Forwards resolution of RHR potential water hammer issue in response to commitment made at enforcement conference re facility operation of RHR sys in suppression pool cooling mode.Informs that util will be revising LER 93-001.

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November 1, 1993
GO2-93-263

Docket No. 50-397

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Gentlemen:

Subject: **WNP-2, OPERATING LICENSE NPF-21**
RESOLUTION OF RESIDUAL HEAT REMOVAL (RHR)
POTENTIAL WATER HAMMER ISSUE

Please find attached our resolution of the subject issue. This is in response to the commitment we made at the recent enforcement conference regarding WNP-2's operation of the RHR System in the Suppression Pool Cooling (SPC) mode.

Consistent with our position taken at the enforcement conference, a LOOP/LOCA accident analysis of RHR in the SPC mode was not included in the design basis accident analysis for WNP-2. This conclusion was justifiable at the time of initial licensing and remains justified notwithstanding current practice regarding the SPC mode. The action taken as indicated in LER 93-001 to consider the associated train of RHR in SPC inoperable and to enter the appropriate Technical Specification Action Statements was, in this light, conservative.

We will be revising LER 93-001 consistent with the position of this submittal.

Sincerely,

RL Webring for

J. V. Parrish (Mail Drop 1023)
Assistant Managing Director, Operations

cc: BH Faulkenberry - NRC RV
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Resolution of RHR Potential Water Hammer Issue

I. INTRODUCTION

This document provides the evaluation which the Washington Public Power Supply System ("Supply System") committed to submit to the Nuclear Regulatory Commission ("NRC") at the conclusion of the recent enforcement conference regarding WNP-2's operation of the residual heat removal ("RHR") system in the suppression pool cooling ("SPC") mode.

The postulated scenario at issue involves a concurrent loss of offsite power ("LOOP")[1] and a loss of coolant accident ("LOCA"), with the initial condition of an RHR train operating in the SPC mode, and is described in Enclosure 1. We have determined, for the reasons discussed below, that when the plant was licensed, the Supply System was not required to consider the possible adverse effects of an accident involving a LOOP and LOCA when RHR was in the SPC mode.

The undocumented reasoning underlying this design basis, apparently accepted at the time, was that -- given presumed use of RHR in the SPC mode -- a simultaneous LOOP/LOCA was not a credible scenario. Moreover, the reasoning which supported the exclusion of that scenario supports the continued exclusion of that scenario even with WNP-2's increased use (relative to that originally expected) of RHR in the SPC mode. Current NRC guidance also supports the exclusion of this scenario as a design basis accident under limits currently placed on the use of RHR in the SPC mode.

Nevertheless, the Supply System recognizes the concern raised by the possibility of a water hammer should a LOCA occur while RHR is in the SPC mode combined with a failure to fast transfer from the Normal transformer to the Startup (230 KV) transformer as described in LER 93-001. A LOCA while RHR is in the SPC mode would cause the RHR system to realign to the Low Pressure Coolant Injection ("LPCI") mode. This alignment when combined with even a short power failure would allow partial RHR system drain down and could result in a RHR system water hammer. The Supply System has determined the extent to which RHR can be used in the SPC mode consistent with the design basis limitation which permits such use, has adopted procedures to ensure that use of the RHR in the SPC mode will not exceed acceptable time limits, and has taken specific actions to reduce the need for SPC mode operation. These actions will ensure that the plant will continue to operate within its design basis assumptions.

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[1] WPPSS-FTS-133, WNP-2 Individual Plant Examination, Main Report; Volume 3, Section AC, Fault Tree Page EAC 8, Cell 75; Aug 1992 (Failure Probability for Fast Transfer from 25 KV Supply to 230 KV Supply)

II. DISCUSSION

A. Initial Licensing Basis

1. Design Basis Assumptions

A thorough search of the Licensing Basis Documents (LBD) for WNP-2 revealed no design basis accident analysis documentation associated with use of RHR in the SPC mode. This is consistent with recent experience with plants of the vintage of WNP-2.

To identify the design basis assumption for use of the SPC mode during normal operation, it has been necessary to rely in substantial part on the institutional memory of General Electric ("GE"), the plant's designer. Based on exhaustive discussions with GE, it was concluded that the sequence of events potentially resulting in a RHR system water hammer event was, at the time of the original design, insufficiently credible to be included in the design basis accident analyses.

Accidents and events that were included in the design basis were documented in the Final Safety Analysis Report ("FSAR") and were analyzed accordingly. This conclusion was based, at least in part, on an assumption that RHR would be used in the SPC mode only a few hours per year during power operation to remove heat introduced to the suppression pool during certain turbine-driven pump surveillance tests. With such limited usage of RHR in the SPC mode, GE made a qualitative determination that the probability of a simultaneous LOOP/LOCA with this initial condition was too low to require consideration. GE has recently documented this reasoning. This approach appears to have been accepted by the NRC at the time the plant was licensed, since the accident analyses were documented in the FSAR.

2. FSAR Description

The FSAR description and analysis of the RHR system at WNP-2 is consistent with GE's understanding of the plant's design basis and shows that operation in the SPC mode was assumed to be highly limited. This would support the reasoning at the time that a LOOP/LOCA at power while in the SPC mode was too improbable to warrant further analysis.

As described in the FSAR, the RHR system can operate in any one of several modes, including SPC and LPCI. The LPCI mode is an engineered safety feature ("ESF") and is actuated by conditions indicating a breach in the reactor coolant pressure boundary.[2] The SPC mode is initiated manually under certain limited conditions; following a LOCA[3] or during normal operation (following RCIC pump surveillance testing).[4]

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[2] WNP-2 FSAR § 1.2.2.5.8.4.

[3] WNP-2 FSAR §§ 3.1.2.4.9.1, 6.2.2.2 and 6.2.2.5.

[4] Id. at § 7.3.1.1.5.b.

The LPCI mode of RHR is an ESF because it is an emergency core cooling system ("ECCS"). Therefore, it meets 10 C.F.R. Part 50, Appendix A, General Design Criterion 35 (GDC 35).[5] WNP-2 emergency core cooling systems are designed to mitigate the consequences of the design basis accidents and events identified in WNP-2 FSAR Chapter 15. Generally, these accidents and events have an estimated frequency of occurrence of greater than once in 10,000 years.[6] The methodology for accident identification has been found acceptable by the NRC Staff.[7]

ECCS initiation signals start LPCI flow.[8] In addition, "the low water level or high drywell pressure signals which automatically initiate the LPCI mode also are used to isolate all other modes of operation and revert system valves to the LPCI lineup."[9] When the initiation signals occur, "the LPCI subsystem has priority through the valve control logic over the other RHR subsystems for containment cooling or shutdown cooling . . . ."[10] "Operator action is not required, except as a monitoring function, during the short term cooling period following the LOCA."[11]

The RHR system functional design in FSAR § 5.4.7.1.1.3 does not discuss any assumptions as to expected SPC mode usage, i.e., hours per year of operation. However, as discussed above, the WNP-2 design basis accident analysis did not include SPC as a starting point for LPCI operation during a LOOP/LOCA apparently because of expected infrequent use assumed for SPC mode during reactor operation at power. During plant design, the SPC mode was seen to be a post-LOCA system, initiated to cool the containment following main steam relief valve ("MSRV") use to lower reactor pressure or as a result of the leak itself. No routine operational need was seen other than to cool the suppression pool following surveillance testing of the turbine driven RCIC pump. This turbine exhausts directly to the suppression pool, an operation which causes substantial heating. During initial design, frequent operation of the SPC mode to remove heat resulting from MSRV steam leakage was not anticipated.

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[5] See Id. at §§ 3.1.2.4.6.1 and 7.3.1.1.1.d.

[6] FSAR § 15.A.3.3.4.

[7] See NUREG-0892, "Safety Evaluation Report Related to the Operation of WPPSS Nuclear Project No. 2, Docket No. 50-397," March 1982, Chapter 15.

[8] Id. at § 6.2.2.3.

[9] Id. at § 6.3.2.2.4. See also Response to Question 211.063. The Susquehanna Units 1 and 2 FSAR description of RHR includes this same language about isolating all other modes of RHR when initiating LPCI.

[10] Id. at § 6.3.3.5. See also § 7.3.1.1.1.4.b.

[11] Id. at § 6.3.3.4.

B. Increased Use of RHR

1. Probabilistic Assessment

As WNP-2 operated, the Supply System found that it was required to use RHR in the SPC mode more than had been recognized initially. This additional use did not initiate any additional concern because WNP-2 had specifically adopted procedures for avoiding the possibility of a water hammer event in the unlikely occurrence of a LOOP/LOCA.

The Supply System did not specifically analyze increased use of RHR. The premise of limited use of RHR underlying the original design basis was, as discussed earlier, based on engineering judgement and, accordingly, there was little focus on those assumptions. However, had WNP-2 evaluated the increased use of RHR, using the tools which were becoming available at the time, it could have concluded, for the reasons discussed below, that even with the increased use of RHR, a LOOP/LOCA with the reactor at power and one train of the RHR system in the SPC mode was still not a credible accident scenario to be included within the design basis accident analyses.

In the late 1970's and early 1980's, as probabilistic risk assessment ("PRA") techniques became more rigorous, GE retrospectively analyzed the various accidents included in boiling water reactor ("BWR") design bases and determined that many had a probability of occurrence of $<10^{-6}$ per year. As a result, GE adopted a design criterion of 10^{-6} events per reactor year as the threshold of credibility for selecting design basis accidents for further analysis.[12] Accidents with a lower frequency were considered incredible and were not analyzed unless specifically required by the NRC.[13]

This conversion of qualitative engineering judgment into quantitative probabilistic analysis is consistent with the NRC Staff's experience during the same time period, as documented in Chapter 3 of draft NUREG/BR-0058, Revision 2, "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission," August 1993:

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[12] GE memorandum OG93-862-01 from S.J. Stark to the BWR Owners Group, September 24, 1993. This criterion was adopted by industry and is set forth in ANSI/ANS-52.1-1983, "Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants." ("[E]vents with a best-estimate frequency of occurrence of $<10^{-6}$/reactor year need not be considered for design. Some events, even though their frequencies of occurrence are $<10^{-6}$/reactor year, have been and may continue to be required to be accommodated within PC-5 nuclear safety criteria by regulatory authorities.")

[13] For example, the LOOP plus LOCA plus single failure, specified by the Standard Review Plan, NUREG-0800, has a probability of occurrence well below 10^{-6}.

In the early development of regulations, this assessment [of the risk of potential changes to public safety] was based on qualitative analysis, simple reliability principles and practices (such as worst case analysis), defense-in-depth and the single failure criterion. The frequency or probability of the hazard was not an explicit factor, primarily because the overall state-of-the-art of [PRA] technology was not sufficiently advanced and accepted.

Applying this quantification of engineering judgment to the current level of use of RHR in the SPC mode, it could be concluded that the initial licensing decision not to consider the LOOP/LOCA with the RHR in the SPC mode remained justified by the low probability of occurrence of that accident sequence. The yearly probability of experiencing the conditions that could lead to a water hammer from this sequence is the product of: (1) the probability of the RHR system being in the SPC mode during normal plant operation, (2) the probability of failure of fast transfer, and (3) the probability of a LOCA large enough to require the LPCI mode. Calculations show that, for the length of time that WNP-2 currently uses RHR in the SPC mode, the probability is less than 10^{-6}/year. Thus, exclusion of this scenario is consistent with the FSAR criterion to exclude accidents and operational transients having a probability of less than once in 10,000 years,[14] and also is consistent with the current GE practice of using 10^{-6} per year.[15]

2. Susquehanna Change of Design Basis

Early in 1987, Pennsylvania Power & Light ("PP&L") informed the NRC of its intention to change the Susquehanna Station's operating design basis for SPC mode of the RHR system as a result of leaking MSRVs.[16] Susquehanna was operating its RHR system in SPC mode "much more frequently" than the 90 hours per year originally assumed. Unable at the time to reduce the need for SPC, PP&L decided to justify the increased operation of the SPC mode.

PP&L concluded that it was safe to operate Susquehanna without a complete assessment of the potential for RHR system water hammer because the combination of events required to support a water hammer occurrence was improbable. Specifically, PP&L determined that the probability of a LOOP/LOCA, assuming continuous operation of SPC, was on the order of 10^{-6} per year. PP&L considered the low probability of a LOOP/LOCA plus the conclusion of NUREG-0927, Revision 1, "Evaluation of Water Hammer Events in Nuclear Power Plants," that

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[14] WNP-2 FSAR §§ 15.A.3.3.3 and 15.A.3.3.4.

[15] GE memorandum OG93-862-01 from S.J. Stark to the BWR Owners Group, September 24, 1993.

[16] PP&L letter from H.W. Keiser to E. Adensam, February 6, 1987.

water hammer damage was limited mostly to pipe supports,[17] to be sufficient justification to support increased SPC usage while PP&L continued to study the event. On the basis of this analysis, Susquehanna modified its FSAR to permit more frequent use of the RHR in the SPC mode.[18] The modification to the FSAR has not been questioned by the NRC.

On December 11, 1986, Susquehanna determined that the potential for a water hammer in the RHR piping was a reportable condition. In response, the NRC issued IE Information Notice No. 87-10: "Potential for Water Hammer During Restart of Residual Heat Removal Pumps" (IN 87-10) early in 1987. That notice corroborates an initial exclusion of the LOOP/LOCA from the design basis accidents required to be considered. The Notice states in part:

> The Susquehanna design basis for LOOP/LOCA assumes that the suppression pool cooling flow path valves are initially closed in the standby lineup. The potential duration factor used in the consideration of the coincident LOCA/LOOP with the RHR in suppression pool cooling mode was one percent, or roughly 90 hours per year. Contrary to the design basis assumption, a licensee review of operating history found that the worst case RHR system usage factor approached 25% during cycles in which significant safety relief valve weepage was experienced.

This initial design basis assumption for Susquehanna is consistent with that made for WNP-2. In addition, this Notice was not followed up with more explicit guidance that would suggest that a design basis deficiency or design defect existed.

Even though no documentation exists for the initial WNP-2 design basis analysis of this issue, it is clear from our discussions with GE that the same analysis was used for WNP-2 as for Susquehanna: RHR usage in the SPC mode was assumed to be so limited that there was no need to consider a LOOP/LOCA under those conditions. Moreover, after the qualitative judgment had been quantified to a yearly probability of less than 10^{-6} as the basis for excluding consideration of an accident, WNP-2 could have determined that it was still within its design basis despite an increase in the use of RHR in the SPC mode, as long as the probability of that configuration and a LOOP/LOCA was less than 10^{-6}/year.

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[17] This may be more true at Susquehanna than at WNP-2. Susquehanna's RHR heat exchangers are at a lower elevation relative to the pump discharge than WNP-2. Thus, the severity of the water hammer is less. See Enclosure 1.

[18] As discussed above, the WNP-2 FSAR did not, and does not contain any reference to assumed use of RHR in the SPC mode. Therefore, for WNP-2 no explicit FSAR change was needed to support "increased" use of that mode.

For distinguishing between RHR system design basis features and other additional features, it is important to note that the NRC Staff did not take exception to the Susquehanna LOCA/LOOP design basis assumption "that the suppression pool cooling flow path valves are initially closed in the standby lineup," and that operation with an SPC usage factor of up to 25% was "contrary to that assumption."[19] In other words, for design basis accident analysis purposes, the SPC mode is not an initial condition for RHR system response to a LOOP/LOCA unless SPC usage factors become excessive.

C. Consistency With Recent Guidance

A LOOP/LOCA while the RHR system is in the SPC mode, even at current levels of use, would not be considered within the WNP-2 design basis by existing guidance.

1. NRC and Industry Guidance

The NRC has issued guidelines[20] (and is about to issue additional guidelines) for evaluating the need for regulatory action to address low probability events, and for deciding whether such events should be further addressed by corrective actions to reduce their safety significance. These guidelines exclude the need for further action to address events having low probability, on the order of 10^{-6}/year (most address a core damage contribution, rather than an event probability). Thus, a threshold for excluding events with probabilities of occurrence of $<10^{-6}$/year from the WNP-2 design basis is consistent with these published and draft NRC guidelines.

The draft NRC Regulatory Analysis Guidelines, applicable to NRC Staff analyses, provide that, if an NRC safety enhancement initiative results in a change of estimated core damage frequency of $<10^{-5}$ per year, the initiative should be terminated unless a strong engineering or qualitative justification dictates otherwise.[21] This guidance is not only consistent with the GE threshold described above, but the GE threshold goes beyond it because the Guidelines address core damage frequency which always is lower than the frequency of occurrence of the initiating event.

Generic Letter 88-20,[22] provides additional guidance indirectly supporting the GE threshold. This Generic Letter endorses the use of PRA as one acceptable means of identifying plant-specific severe (beyond-design-basis) accident vulnerabilities, and provides quantitative

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[19] IN 87-10 at 1.

[20] Generic Letter 88-20.

[21] NUREG/BR-0058 § 3.3.1 We understand that these draft guidelines are, for the most part, approved and should not be altered significantly in their final form.

[22] Generic Letter 88-20, "Individual Plant Examination For Severe Accident Vulnerabilities - 10 C.F.R § 50.54(f)," November 23, 1988.

criteria for identifying accident scenarios that the NRC considers worthy of further evaluation under the IPE program. GL 88-20 specifies that sequences meeting any of these criteria should be reported to the NRC in the IPE submittal. Specifically, Appendix 2 to GL 88-20 provides the following five criteria for reporting of important severe accident sequences:

- any functional sequence that contributes 1×10^{-6} or more per reactor year to core damage;
- any functional sequence that contributes 5% or more to the total core damage frequency;
- any functional sequence that has a core damage frequency greater than or equal to 1×10^{-6} per reactor year and that leads to containment failure which can result in a radioactive release magnitude greater than or equal to the BWR-3 or PWR-4 release categories of WASH-1400;
- functional sequences that contribute to a containment bypass frequency in excess of 1×10^{-7} per reactor year; and
- any functional sequence that the utility determines from previous applicable PRAs or by utility engineering judgment to be an important contributor to core damage frequency or poor containment performance.

However, it is emphasized in Appendix 2 to the Generic Letter that these criteria do not represent a threshold for vulnerability. Thus, event sequences with higher or lower frequencies of occurrence can be included or excluded from reporting given sufficient justification. These guidelines also support, indirectly, the 10^{-6}/year threshold used by GE. In fact, the threshold of 10^{-6} is even more conservative in that it is an event probability, rather than the probability of core damage.

The GE threshold also is consistent with industry guidance for evaluating severe accident (i.e., beyond-design-basis) sequences. NUMARC's Severe Accident Closure Guidelines, NUMARC 91-04, suggest in Section 2.2 that "no further corrective action need be pursued for those sequences with [core damage frequency] values below the threshold of 1E-6 per reactor year (or 1E-7 for containment bypass events)" (tempered by the exercise of judgment based on the consequences). Again, these guidelines are directed to core damage frequencies, not the probability of the sequence itself. Thus, a 10^{-6} threshold is more conservative than these guidelines.

Finally, the GE threshold is consistent with industry standards; specifically, ANSI/ANS-52.1-1983 ("[E]vents with a best-estimate frequency of occurrence of $< 10^{-6}$/reactor year need not be considered for design. Some events, even though their frequencies of occurrence are $< 10^{-6}$/reactor year, have been and may continue to be required to be accommodated within PC-5 nuclear safety criteria by regulatory authorities.").

2. Operation Limits Keep the Probability of a RHR Water Hammer Event Below Design Basis Limits

The Supply System currently limits operation of the RHR in the SPC mode to ensure that the usage remains within current assumptions. The RHR system is operated in the SPC mode no more than an average of 15 hours per week.[23] This limited operation provides ample margin to the design basis limit.

Assuming one train of RHR is operated 15 hours per week in SPC mode,[24] the probability of a severe water hammer from a LOOP/LOCA has been estimated to be 2.9×10^{-7}/year. This probability estimate assumes all break sizes, even though not all LOCAs will require LPCI operation, and the containment pressures resulting from the largest LOCAs will prevent voiding, and subsequent water hammer, in the RHR system. If only LOCAs which require LPCI operation are considered, the probability is reduced by a factor of 20.[25]

Not only is this outside the GE threshold criterion of 10^{-6} per event per year, it also is outside the current NRC guidelines for taking actions to address accident sequences outside a facility's design basis. As discussed above, the NRC Regulatory Analysis Guidelines suggest terminating safety enhancement initiatives where the change in core damage frequency is $<10^{-5}$ per year. The WNP-2 event probability is well below this value; core damage frequency should be even lower.

Generic Letter 88-20 Sequence Selection Criteria suggest that this event would not require reporting if discovered during the IPE process. Not only is the event frequency well below the 10^{-6} criterion, but the effect on WNP-2 core damage frequency, (5.42×10^{-5} per reactor year), is less than a 0.54 percent increase. This event also would not require action under the IPE program according to NUMARC's Severe Accident Closure Guidelines, NUMARC 91-04. As discussed above, these guidelines suggest a core damage frequency threshold of 10^{-6} per reactor year. The calculated event frequency of about 10^{-7} per year would result in a core damage frequency well below the NUMARC 91-04 threshold.

Finally, the 10^{-7} per year RHR water hammer event frequency is below the 10^{-6} threshold in ANSI/ANS-52.1-1983; therefore, the event may be excluded from further evaluation consistent with industry guidelines.

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[23] Attachment 1 to the Basis For Continued Operation for PER 292-1243; WNP-2 Licensee Event Report No. 93-001-01, "Further Evaluation and Corrective Action" paragraph C.3.

[24] This is well below the 25% noted by the NRC Staff for Susquehanna in IN 87-10.

[25] Attachment 1 to BCO for PER 292-1243; LER 93-001-01, "Further Evaluation and Corrective Action" paragraph C.3; WPPSS Interoffice Memorandum from A.J. Moore to D.W. Coleman, August 30, 1993.

D. Ongoing and Proposed Enhancements

The Supply System has previously taken actions to minimize the probability and consequences of water hammer events. The Supply System is still aggressively pursuing the matter. The Supply System has taken or is studying actions to reduce SPC mode usage and to minimize the potential for RHR water hammer as documented in LER 93-001.

III. CONCLUSION

An accident sequence for a LOOP coincident with a LOCA, starting with RHR in the SPC mode, was not in the original WNP-2 design basis. An assumption of low use of the SPC mode led to the conclusion that the accident sequence was a low probability scenario that need not be analyzed. The assumption was later quantified.

The Supply System has increased SPC use beyond that originally expected. However, based on the original assumption as to what scenarios needed to be included in the design basis accident analysis, and the subsequently quantified probabilistic threshold, the increased usage does not warrant new or changed design basis accident analyses; the postulated scenario remains a very low likelihood event.

Precautions have been taken such that increased use of RHR for SPC is limited and remains within these assumptions. Moreover, under current NRC guidelines[26], current use of RHR in the SPC mode would not have resulted in a requirement to include the RHR water hammer accident in the WNP-2 design basis.

[26] NUREG/BR-0058.

ENCLOSURE 1

Description of Postulated RHR Water Hammer Event

At WNP-2, the RHR system provides a number of functions including SPC and LPCI. When in the SPC mode, RHR pumps remove water from the suppression pool, at the 435 foot elevation, and pump it to heat exchanges at about the 586 foot elevation. From the heat exchangers, water flows back to the suppression pool, below the pool surface, at the 466 foot elevation.

The following sequence of events can occur following a loss of offsite power (LOOP) with a coincident loss of coolant accident (LOCA) while an RHR train is operating in the SPC mode (starting at time = 0 sec.):

Time	Action
0	RHR pumps coast down as a result of the LOOP;
0+	emergency diesel generators (EDGs) start; as RHR pumps coast down water drains from the heat exchanger discharge line into the suppression pool as water in the higher elevations of the RHR pipe flashes and forms a void;
10	EDGs reenergize the emergency power buses;
10+	RHR motor operated valves begin to realign (and continue realigning for 30 - 40 sec.);
15	RHR pump starts;
17	RHR flow reaches maximum velocity;
17+	Water hammer occurs in RHR heat exchanger, at an estimated 130,000 lbf, possibly damaging heat exchanger baffle plates and nozzles; the reflected pressure pulse would significantly exceed the allowable stress limit of the RHR pump inlet nozzle and mounting.