Docket File

UNITED STATES
NUCLEAR REGULATORY COMMISSION
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April 21, 1995

Mr. B. Ralph Sylvia
Executive Vice President, Nuclear
Niagara Mohawk Power Corporation
Nine Mile Point Nuclear Station
P.O. Box 63
Lycoming, NY 13093

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION REGARDING INDIVIDUAL PLANT EXAMINATION (IPE) FOR NINE MILE POINT NUCLEAR STATION UNIT NO. 1 (NMP-1) (TAC NO. M74436)

Dear Mr. Sylvia:

By letter dated July 27, 1993, Niagara Mohawk Power Corporation (NMPC), submitted its response to Generic Letter 88-20, Supplement 1, "Individual Plant Examination for Severe Accident Vulnerabilities - 10 CFR 50.54(f)," for NMP-1.

The NRC staff has begun its review of NMPC's July 27, 1993, submittal. However, we have determined that additional information, as identified in the enclosure, is required to complete our review of the submittal. As indicated in the attached request for additional information (RAI), additional information regarding the internal event analysis of the IPE, including front-end, back-end and human reliability portions, and the containment performance improvement program is required. NMPC is requested to respond to this RAI within 60 days of receipt of this letter in order for us to complete our review in a timely manner.

This requirement affects nine or fewer respondents and, therefore, is not subject to the Office of Management and Budget review under P.L. 96-511.

Sincerely,

for

Gordon E. Edison, Senior Project Manager
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Office of Nuclear Reactor Regulation

Docket No. 50-220

Enclosure: Request for Additional Information

cc w/encl: See next page

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REQUEST FOR ADDITIONAL INFORMATION

REGARDING INDIVIDUAL PLANT EXAMINATION (IPE)

NIAGARA MOHAWK POWER CORPORATION

NINE MILE POINT NUCLEAR STATION UNIT NO. 1

DOCKET NO. 50-220

1. Provide the plant freeze date used in the IPE and discuss any plant modifications or changes in procedures, pending as of the freeze date, that were considered in the IPE models.

2. Explain how the following initiating events either were screened from consideration as plant specific initiating events or were lumped into higher frequency initiating event categories.

   (a) Loss of heating, ventilation, and air conditioning (HVAC) is mentioned in Table 3.1.1.5; however, it is not clear how it was treated as part of the plant specific initiating event analysis. Provide a brief and concise discussion of loss of HVAC (e.g., for control room cooling) as an initiating event. If credit was taken for controlled plant shutdown or for compensatory cooling options explain how loss of HVAC is detected in time to allow such actions.

   (b) Table 3.1.1.5 states that loss of a single DC battery board should not cause a plant trip. Explain whether an operator action is required to prevent plant trip following loss of a DC battery board. If operator action is needed, describe the action and provide its human reliability analysis (HRA) evaluation. Include in the discussion consideration of the time available and time needed to perform the action and how these times were incorporated in the HRA.

3. The submittal states that recirculation pump seal loss-of-coolant accident (LOCA) initiating events can be lumped in with a small LOCA. However, the small LOCA frequency used in the IPE is about an order of magnitude lower than seal LOCA initiating event frequencies typically seen in probabilistic risk assessments (e.g., NUREG-1150). Provide the basis for the frequency used to quantify a seal LOCA initiating event.

4. The submittal's success criteria for manual depressurization is the opening of two relief valves. The updated final safety analysis report (UFSAR) criterion though is the opening of three relief valves [UFSAR, Page V-17 and Table XV-9] which is in agreement with the criteria typically used in boiling-water reactors (BWRs), for example, in the NUREG-1150 (Peach Bottom and Grand Gulf). The concern is that, in order to achieve depressurization with two instead of three relief valves it may require significantly greater time; this may result in core uncovery

Enclosure

for longer than anticipated time, and possibly, greater than the typically assumed 3 feet from the bottom of the active fuel; therefore, it may cause core damage even with successful low pressure coolant injection.

(a) Please provide the basis for the assumption that two relief valves are adequate for successful depressurization.

(b) The IPE credits fire water as one of the low pressure systems used in conjunction with depressurization. Typically, the firewater system is not credited as a low pressure cooling injection system during depressurization, but as a backup system to the emergency core cooling system (ECCS) low pressure systems. That is, the firewater is typically modeled as injecting into the reactor after the vessel pressure has reached considerably lower levels by means of ECCS low pressure injection. Please provide the basis for assuming that firewater can be used in conjunction with depressurization; include in your discussion the maximum vessel pressure at which firewater can inject.

5. The IPE assumes that core damage will likely not occur prior to containment failure and that low pressure injection systems can continue cooling the core while both containment cooling and containment venting have failed. Specifically, the IPE assumes: (1) a core spray pump with design temperature of 140 °F can endure temperatures up to about 310 °F (corresponding to containment failure) with a probability of 0.5, and (2) upon termination of injection from external water sources and loss of core spray due to high temperature, containment failure will precede core damage with a probability of 0.8. Hence, it is assumed that 90% of the time core damage will not occur prior to containment failure, and therefore, the IPE takes credit for low pressure cooling in sequences where both containment cooling and venting have failed.

(a) Provide more technical information supporting the assignment of the two probabilities: 0.5 and 0.8.

(b) Discuss how sensitive the overall core damage frequency (CDF) is to the numerical values of these two probabilities.

6. A seal LOCA is defined as resulting in leakage greater than 45 gpm after 1 hour. The probability of a seal LOCA with loss of seal cooling is taken to be 0.05, crediting the cooldown afforded by the isolation condensers which reduces the likelihood of seal failure. This means that a transient with loss of seal cooling most likely does not involve a seal LOCA. However, leakage rates up to the seal LOCA limit can result in loss of adequate vessel inventory over the 24-hour mission time, especially when shrinkage of the vessel water due to cooling with the isolation condensers is considered. Based on the station blackout (SBO) event tree and the discussion of the dominant core damage sequences it appears that makeup is required. Based on the general transient event

tree, however, it appears that makeup is not required upon loss of seal cooling. Therefore, it is not clear how the IPE modeled the requirement for makeup to the vessel especially when cooling is accomplished with the isolation condensers.

(a) Clarify what requirements for reactor coolant makeup were modeled when long term reactor cooling is provided by the isolation condenser during: (1) SBO and (2) a general transient with seal cooling lost.

(b) The success criteria table indicates that during a SBO leakage up to 25 gpm can be tolerated without makeup. Explain how core cooling with isolation condenser is maintained over the 24-hour mission time with so much leakage especially when primary coolant shrinkage due to cool down is taken into consideration.

7. Address the following issues associated with system modeling:

(a) Clarify whether the IPE assumed, as indicated in Section 3.2.1-23 of the submittal, that no HVAC nor ventilation systems are required to support the operation of frontline systems. If they are required, discuss their treatment in the IPE.

(b) The submittal states that HVAC for the DG rooms is not required because the rollup doors are normally partially open in hot weather. Explain: whether the opening of the rollup doors is required by procedure; what is the probability that the doors are not open, (because, for example, of cold weather); and, how it was taken into consideration.

8. Address the following questions related to data used in the IPE:

(a) Provide the process by which the components that were quantified with plant-specific data were selected.

(b) It appears that a single failure rate (a failure rate for a 'general' pump) was used for all types of pumps as, for example, the core spray, containment spray, and containment spray raw water pumps. Typically, however, different pump failure rates are estimated based on plant-specific data for different pump types. Especially care is taken to differentiate between the data used to estimate failure rates for the high-pressure pumps from the data used to estimate failure rates for the low-pressure pumps. Explain how the failure rates for the different pump types were derived; in particular discuss the estimation of failure rates for the core spray, containment spray, and containment spray raw water pumps.

(c) It appears that the same common cause failure (CCF) factors were used for all centrifugal pumps. In addition, these ("general" pump) factors appear to be low (order of magnitude) as compared to other

data for similar plants. (1) Please explain how the CCF factors were estimated and explain how these low CCF factors did not mask any potential vulnerabilities; and (2) please clarify whether the same CCF factor was used for all pumps. If the same CCF factor was used, please provide the basis for this assumption.

(d) Provide the basis for the relatively low beta factors used for (e.1) diesel generators and (e.2) relief valves.

9. While the process used to identify and select miscalibration errors from the instrument loop functions is discussed, it is not clear from the submittal what was the basis for selecting the risk important instrument loop functions. Explain how instrument loops and functions were selected ensuring that important miscalibration events would not be eliminated.

10. The submittal identified an important preinitiator event: calibration of the core spray low pressure permissive (ZP301). The submittal, however, did not provide a discussion of how the human error probability (HEP) of 1.558 E-4 (Table 3.4.2-4) was derived. Discuss the derivation of this value. Include in your discussion the method used and the consideration of plant specific factors; i.e., performance shaping factors and recovery factors.

11. The submittal provides a general discussion of the process used for estimating HEPs for post-initiator events. It does not, however, provide specific information illustrating this quantification process. For example, it does not provide the basic human error probabilities (BHEP) and the modification factors used to derive these HEPs. For each of the actions: (i) inhibit automatic depressurization system (ADS) during anticipated transient without scram (ATWS), (ii) shed loads of DGs during LOCA; and (ii) depressurize during transients with loss of high pressure makeup:

(a) Distinguish the detection-diagnosis-decision type error from the execution type error and demonstrate how the HEPs associated with each type error was calculated. In the discussion (a.1) identify the BHEP values used along with their sources; (a.2) where applicable, provide the factors used to modify these BHEPs along with the basis for choosing these values; (a.3) provide the time "required" (for either detection-diagnosis-decision or execution) and explain how these times were measured (e.g., simulator exercises, walkdowns, etc).

(b) Explain how the HEP associated with each type error in (a) was combined to derive a final HEP for the action.

(c) Illustrate how different HEPs were estimated for the same action under different accident sequences.

12. Provide the percentage contribution of the sequences with the containment vented to the total CDF, independent of whether containment fails later by some other mechanism.

13. Tables 4.4-6 through 4.4-14 imply that containment will fail at the vent line bellows roughly 99% of the time (the only exception being Tables 4.4-12 and 4.4-13 where the temperature is greater than 800 °F and the failure is in the drywell shell and head seals). However, Figure 4.6-17 implies that 42.7% of the time (50% of the releases) the drywell shell fails, among other failure modes. Provide a discussion that reconciles this seeming discrepancy. As part of the discussion, specify the percentage of the total CDF of drywell failure by shell melt-through and by other means, and of wetwell failure.

14. It is not clear from the submittal whether Revision 4 of the emergency procedure guidelines (EPGs) have been incorporated into the emergency operating procedures (EOPs), as recommended by the CPI program. Please explain whether Revision 4 of the EPGs were used for the IPE or provide a status regarding their incorporation.

15. Explain the relationship between the various split fractions for the containment event tree (CET) top events and the severe accident and containment conditions. For example, the early containment failure top event "CZ" in CET1 and CET2 has 11 split fractions with different values (i.e., "CZ1" through "CZ10" and "CZF" as summarized in Table 3.3.5-1 on Page 3.3.5-4). What accident and containment conditions do these split fractions represent and to which accident sequences do they belong? Provide an example. Also, explain why the split fractions "CZ3" and "CZ4," which have identical logical descriptions (rules), have different probabilities.

16. Figure 4.6-17 (Page 4.6-58) shows that 15% of the total release would result from energetic failure and another 5% from energetic failure during internal flooding. Energetic failures postulated in the submittal (steam explosions, direct containment heating, hydrogen combustion, and overpressure at reactor pressure vessel failure) were addressed in the CET node CZ/CE. Because of the importance of this information for setting accident management initiatives, describe your quantification of this node given in Table 3.3.5-1.

17. The submittal notes that a containment melt-through with an area of $2ft^2$ is sometimes followed by a drywell head seal failure (Page 4.6-19 and Figure 4.6-17). This suggests a containment pressure buildup even after containment melt-through, which in turn indicates that the containment melt-through would not result in containment pressure relief. Under such circumstances, it is not clear how containment melt-through by itself is considered to be a release path (9% of the total release as given in Figure 4.6-17). Please clarify.

18. Table 4.5-3 (Page 4.5-21) notes that invessel recovery or arrest of core melt progression invessel could be accomplished using "makeup systems (identified in the EOPs) with capacity greater than approximately 1000 gpm." Because of the many different split fraction values used for the CET node IR/RX, the invessel recovery potential is not clear. What was the total conditional probability of invessel recovery, given core damage?

19. According to page 4.9-32 of the submittal, it appears that two sensitivity cases were run for the node CZ/CE, one by assuming an optimistic value of "ϵ" (close to zero) and another by assuming a pessimistic value of 1.0. However, we were not able to find the descriptions of these two sensitivity cases in the submittal. Please provide these descriptions.

20. The Nine Mile Point Unit 1 (NMP-1) drywell sumps have the capability to contain about 45% of the total corium inventory, greater than typically seen in other BWRs with Mark I containments. In addition, the drywell has five pedestal openings, allowing for a more even spread of the remaining corium on the drywell floor (relative to other Mark Is). For these two reasons, one would expect less corium coming into contact with the drywell shell and a lower resulting probability of liner melt-through (page 8-10). Relative to other failure mechanisms, one would expect that drywell melt-through would not be an important contributor to the total release. However, the IPE's drywell melt-through release of 2.3×10^{-6} per year is 50% of the total release. It is not clear what drives this drywell melt-through contribution. Please explain.

21. Table 5-6 in Section 5.3 of the submittal is missing. Provide a copy of the table.

22. Figure 4.6-18 (Page 4.6-59) indicates that 38% of the "Large" release comes from the "Overpressure Wetwell" failure. However, it is not clear what are the pathways that lead to overpressure wetwell failure. Please describe the various pathways that contribute to this release.

23. Figure 4.6-17 (Page 4.6-58) indicates that 41% of the total release comes from the "Shell and DW Head" failure. It is our understanding that the shell fails first, but with a failure size insufficient to depressurize the drywell. Subsequently, through pressure and temperature increases, the drywell head fails. Is this understanding correct? In this figure, what is the contribution from containment melt-through? Is it the 9% "Shell Failure?" In any of the sequences, does the drywell shell fail by over pressurization. If so, describe.

24. The Level 2 analysis includes credits for operator actions by following the EOPs. The individual actions and the negative aspects of some of those actions are well described. Since the human actions may have both positive and negative aspects, the overall impact of these actions should be examined.

(a) Provide the characterization of the releases (release categories) without these credits in the Level 2 portion of the accidents.

(b) Explain how would Figures 4.6-17 and 4.6-18 change without the credit for human actions.

25. Section 6.0 (Page 6-1) states that "Section 6.2 focuses on those improvements judged to have the most potential benefit in reducing ... risks." Section 6.2 (Page 6-5) states "that any plant improvement aimed at one specific area (or sequence type) and based purely on the IPE results may not be cost beneficial." Section 6.5 goes on to say that the potential improvement with the greatest possibility to reduce risk is associated with SBO, namely the need for AC and DC power. Specifically, DC is needed in order to restore AC power. Two improvements are mentioned: "procedural improvements that shed the nonsafety battery such that it would be available as a backup" and a portable generator. The submittal is not clear as to whether either or both of these SBO related potential improvements have or are going to be incorporated into the NMP-plant. Since the reactor pressure vessel relief valves are dependent solely on DC power the importance of enhanced DC power is the dominant issue related to the CPI recommendation for NMP-1.

In addition, the discussion of the importance of the four recovery actions identified in Level 2 analysis, can be interpreted as the need for improvements in operator training with regard to these scenarios was deemed warranted by the analyst. The submittal is not clear as to whether this particular improvement or any other front-end or back-end related improvement has been or will be implemented.

(a) Provide a discussion which clarifies which potential improvements discussed in Section 6.2 have been (or are going to be) made at NMP-1. Include in your discussion their impact on the CDF or radioactive release levels.

(b) In particular, discuss your evaluation of the CPI recommendation related to enhanced reactor depressurization capability with the rationale for making (or not making) enhancements to the plant and procedures.

26. It is not clear in the submittal what plant changes due to the SBO rule were credited in the analysis. Please provide the following: (1) identify whether plant changes (e.g., procedures for load shedding, alternate AC power) made in response to the blackout rule were credited in the IPE and what are the specific plant changes that were credited; (2) if available, identify the total impact of these plant changes to the total plant CDF and to the SBO CDF (i.e., reduction in total plant CDF and SBO CDF); (3) if available, identify the impact of each individual plant change to the total plant's CDF and to the SBO's CDF (i.e., reduction in total plant CDF and SBO CDF); (4) identify any other changes to the plant that have been implemented or planned to be implemented that

are separate from those in response to the SBO rule, that reduce the SBO's CDF; (5) identify whether the changes in #4 are implemented or planned; (6) identify whether credit was taken for the changes in #4 in the IPE; and (7) if available, identify the impact of the changes in #4 to the SBO's CDF.

April 21, 1995

Mr. B. Ralph Sylvia
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SUBJECT: REQUEST FOR ADDITIONAL INFORMATION REGARDING INDIVIDUAL PLANT EXAMINATION (IPE) FOR NINE MILE POINT NUCLEAR STATION UNIT NO. 1 (NMP-1) (TAC NO. M74436)

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Sincerely,

Original signed by:

Gordon E. Edison, Senior Project Manager
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Docket No. 50-220

Enclosure: Request for Additional Information

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