From: <leonardm@nimo.com>
To: WND2.WNP3(DSH)
Date: 5/22/98 1:32pm
Subject: Open Questions - U1 CREVS

Darl, attched is a list of open questions as I understand them. Please review and confirm this is your understanding. NMPC requests that (*) questions be reconsidered because they may not necessarily be needed to support the amendment. How many of the others need to be docketed?(See attached file: NRCQUESTIONS.doc)

date of or source of	*question*
RAI QUESTION 1	are there accidents for which the radiation monitors are depended upon to INITIATE the CONTROL room system to ensure that the control room operator doses are within the limits of gdc 19?
rai * QUESTION 1	is the issue with the control room intake radiation monitor a design problem or a TECHNICAL specificatio
RAI * QUESTION 1	IF A NEW RADIATION MONITOR WERE INSTALLED IN THE CONTROL ROOM INTAKE, WOUL SENSITIVITY PROBLEM BE SOLVED?
RAI QUESTION 5/8	WHAT IS THE JUSTIFICATION FOR ASSUMING THAT THE MSIV LEAKAGE AND ECCS LEAKA THE STACK VERSUS THE TURBINE BUILDING VENTS OR SOME OTHER PATHWAYS? PROVI VENTILATION SYSTEM AND DUCT FLOW PATH.
RAI QUESTION 5/8	BASED ON THE HISTORICAL DATA, IT WOULD SEEM APPROPRIATE TO ASSUME THAT THE SCFH(?)
5/20/98	I [DARL HOOD] ALSO THOUGHT THAT A REVISED UNIT 1 MSLB ANALYSIS WAS TO BE PERF QUANTITY OF STEAM RELEASED DURING HOT STANDBY WAS GREATER THAN AT FULL P HAD BEEN PERFORMED ASSUMING FULL POWER(?)
5/21/98	THE FOLLOWING INFORMATION IS NEEDED: THE NUMBER OF FUEL RODS IN THE UNIT 2 CORE. THE CONDENSER FREE AIR VOLUME FOR BOTH UNITS 1 AND 2.
5/19/98	IS THE CONTROL ROOM EQUIPPED WITH AREA RADIATION MONITORS?
5/21/98	I [JOHN HAYES] QUICKLY PERUSED THE LICENSEE'S CALCULATIONS AND SAW THAT THE RECTOR COOLANT ACTIVITY MUST BE LIMITED TO A TOTAL IODINE LEVEL OF 9.47 uCi/g I DOSE TO 30 REM THYROID. CURRENT TECHNICAL SPECIFICATION VALUE IS 25. IS NINE MI TECHNICAL SPECIFICATION CHANGE OR ARE THEY GOING TO DEPEND UPON THE RADIAT TO INFORM US OF THEIR PLANS.

* NMPC proposes that these answers are not required

From: John Hayes
To: WNP3.DSH
Date: 5/20/98 8:42am
Subject: review of nine mile unit 1 rai responses

I have reviewed the draft responses and have provided the following for you to forward to the licensee. I am having the containment systems people review the MSIV leakage issue.

Jack

<WP Attachment Enclosed>

CC: rle

I have the following comments with respect to Nine Mile Unit 1's response to the NRC's request for additional information.

Question 1

The question really applied not only to the rod drop but also the SBLOCA and accidents originating from Nine Mile Unit 2 or Fitzpatrick. The basic question is, "Are there accidents for which the radiation monitors are depended upon to initiate the control room ventilation emergency ventilation filtration system to ensure that the control room operator doses are within the limits of GDC 19?"

Is the issue with the control room intake radiation monitor a design problem or a technical specification issue?

If a new radiation monitor were installed in the control room intake, would the monitor's sensitivity problem be solved?

Question 5/8

It is unclear that either the licensee or the staff addressed ECCS leakage and/or MSIV leakage. The staff's workbook on the LOCA doses indicates that only containment leakage at 1.9%/day was considered in the analysis. There needs to be a basis fro the assumptions which form the input for your accident dose calculations. The staff's question is in essence, "What is the justification for assuming that the MSIV leakage and ECCS leakage will be released by the stack versus the turbine building vents or some other pathway?" There is nothing in the UFSAR that discusses the release points associated with these radiation sources. This information should be available. There is a technical answer to this question.

Based upon the historical data, it would seem inappropriate to assume that the MSIV leakage totals 8 scfh.

DRAFT

NRC Request for Additional Information and
Niagara Mohawk Power Corporation (NMPC) Responses

1. *[Provide the basis for not] Maintaining the Hi-rad initiation signal in TS Table 3.6.2l of the technical specifications in order to ensure that control room ventilation ESF system is initiated to mitigate the consequences of a rod drop accident.*

Question applies to rod drop, SBLOCA, and Unit 2 and Fitzpatrick accident

NMPC Response:

The control room ventilation radiation monitors are not needed to mitigate the consequences of a control rod drop accident (CRDA) because the control room emergency ventilation system (CREVS) is not required to mitigate a CRDA.

At NMP1 the severity of a CRDA is reduced by strict procedural controls, supplemented by use of a rod worth minimizer (RWM). Control rod withdrawal and insertion sequences are established to assure that the maximum in-sequence individual control rod or control rod segments which are withdrawn could not be worth enough to cause the core to be more than 0.013 delta-k supercritical if a rod drop accident were to occur. The severity is further reduced by limiting the maximum "dropout velocity" of any control rod with the rod velocity limiter.

If the radiation resulting from the CRDA excursion is not intensive enough to cause automatic isolation of the offgas piping and the Operator fails to isolate the piping manually, the NMP1 design is such that the fission products are released to the stack after a 30-min delay. In this case, even using conservative meteorological assumptions, operator doses are well below 10CFR50, Appendix A, GDC 19 criteria.

The NMP1 CREVS radiation monitors are in-line gas monitors. The required setpoint of the monitors is so close to the monitor sensitivity that spurious actuation may occur. For postulated accidents, except the main steam line break (MSLB), the monitors cannot be calibrated to provide proper response because of this problem. For this reason, NMPC has proposed to initiate the CREVS using RPS signals.

Question - Would a new radiation monitor fix the monitor sensitivity problem?

Issue is whether the radiation monitor setpoint is a TS issue or a design issue

OK 2. *[Provide a discussion of] Whether the proposed changes sufficiently address a small break LOCA.*

<u>NMPC Response:</u>

NMPC has reconsidered its position that Small Break LOCAs outside of the primary containment will not be analyzed for control room habitability. We have reviewed the Staff's position that NUREG-0737 TMI Task Action Plan, Item III.D.3.4, "Control-Room Habitability Requirements," required all design basis accidents to be evaluated to determine if the radiological consequences might constitute a greater hazard to control room habitability than the design basis LOCA. According to the Staff, the evaluation should have included Small Break LOCAs outside of containment. The only line breaks outside of containment discussed in the NMP1 UFSAR are the feedwater line break and the MSLB. However, we believe it prudent to evaluate, for the purposes of control room habitability, a limited number of additional line breaks which could be postulated to occur outside of containment. Specifically, NMPC has performed additional evaluations of 1) an Emergency Condenser line break, 2) an instrument line break, 3) a Reactor Water Cleanup line break, and 4) a Shutdown Cooling system line break. Based on analysis of these four line breaks, operation of the Control Room Air Treatment System is not required to mitigate their effects in order to meet the GDC 19 dose limits for the control room. The calculations confirming these results will be transmitted for your review under a separate cover letter.

OK 3. *[Provide a] Reference to the dose assessment or transmittal of the dose assessment which indicates operation of the control room ESF ventilation system is required in the event of a Fuel Handling Accident.*

<u>NMPC Response:</u>

The dose assessment calculation for the Fuel Handling Accident will be transmitted for your review under a separate cover letter.

OK 4. *[Provide the] MSLB and LOCA control room dose calculations.*

<u>NMPC Response:</u>

The dose assessment calculations for the MSLB and LOCA will be transmitted for your review under a separate cover letter.

It is not clear that the

5. *[Provide a discussion of] Whether ECCS leakage and MSIV leakage during a LOCA is processed by the SGTS and released via the elevated stack.*

NMPC Response:

Stack release of post-LOCA bypass leakage from the MSIVs and the ECCS is discussed in the NMP1 license basis and in docketed communications with the Advisory Committee on Reactor Safeguards (ACRS), the Atomic Energy Commission (AEC), and the NRC.

Question

Both the Full Term Operating License Technical Specification (TS) Bases and the current TS Bases for the primary containment leakage rate (TS 3.3.3 and 4.3.3) describe the LOCA analysis as follows: "The design basis loss-of-coolant accident was evaluated at the primary containment maximum allowable accident leak rate of 1.9%/day at 35 psig...Thus, the doses reported are the maximum that would be expected in the unlikely event of a design basis loss-of-coolant accident. These doses are also based on the assumption of no holdup in the secondary containment resulting in a direct release of fission products from the primary containment through the stack to the environs." The LOCA release pathway was assumed to be from the stack only. This release path was acknowledged and accepted by the ACRS as documented in their Safety Evaluation for NMP1, dated March 24, 1969. The Safety Evaluation states that: "Release from the 350-foot stack was assumed for all accidents except the steam-line-break and control-rod-drop accidents..." The AEC provided confirmation of the analysis release path in an April 23, 1969 letter, which states that: "Credit for release of activity from the 350-foot stack was given except for the steamline-break and control-rod-drop accidents."

NMPC's letter (T.E. Lempges to D.B. Vassalo), dated March 1984, provided a Control Room Habitability Report to the NRC in response to TMI Action Item III.D.3.4. The report addressed control room operator radiation exposures resulting from design basis accidents and states, in part, "In addition, for conservatism, eight valves in the Turbine Building are assumed to leak depressurized coolant at a rate of one standard cubic foot per hour each. This release goes to the stack unfiltered." Based on this analysis and committed modifications that were subsequently implemented, the NRC issued a Safety Evaluation, dated May 21, 1984, which concluded that the control room dose analyses were acceptable.

OK 6. *Explain why the LOCA analysis assumes a leakage rate of 1.1% by weight of containment air per day when the TS allowable leakage rate (<1.5% by weight) could be greater.*

NMPC Response:

At NMP1, the primary containment post accident leakage rate of 1.9 weight percent per day at 35 psig is the allowable leakage consisting of steam, moisture, water vapor, noble gasses, nitrogen, hot air devoid of oxygen, et. al. at 35 psig. The current TS Bases state that the 1.5 weight percent per day at 35 psig (LA) is a ratio (0.8) of the 1.9 weight percent per day at 35 psig of the post accident atmosphere compared to the test condition atmosphere (dry-air mass). Until 1997, NMP1 conducted the Containment Integrated Leak Rate Test (CILRT) at the reduced pressure of 22 psig.

NUREG 0737 states that Standard Review Plan (SRP) 15.6.5 should be used to analyze the radiological consequences for control room habitability. SRP 15.6.5 states that "The leakage rate used should correspond to that given in the technical specifications." In 1984, the TS Maximum Allowable Leak Rate was 1.1 weight percent per day at 22 psig [LT (22)]. LT (22) is equivalent to LA based on the dry-air mass of the containment, corrected for the difference in pressures. The term LA was not defined in the TS in 1984.

In 1998, the more appropriate term for use in the radiological calculations is LA (1.5 weight percent per day at 35 psig); which is the current TS Maximum Allowable Leak Rate, although LA and LT are equivalent. The dose assessment calculation for the LOCA that will be transmitted for your review per Item 4 above will use LA.

7. *What is the basis for the MSIV leakage number?*

NMPC Response:

Leakage is calculated using the latest 10CFR50 Appendix J Types B and C test results for selected penetrations. The penetrations used are the MSIV lines (2), the Feedwater Lines (2), the Emergency Cooling Steam Supply and Water Return Lines (4), the Drywell Vent Line (1) and Torus Vent Line (1), and the Main Steam and Feedwater penetration bellows assemblies. The Vent Lines are included, since the capability exists to vent the lines to the main condenser.

The administrative limit for leakage was established based on a radiological calculation to ensure control room habitability. The leakage limit is presently 42.000 SCFH @ Pa (35 psig) with an individual valve Inservice Test (IST) and 10CFR50 Appendix J Testing Program Plan limit of 32.3 SCFH @ Pa. Secondary Containment Bypass Leakage is not an NMP1 Technical Specification requirement.

8. *Provide the basis for the assumption that MSIV leakage is released via the stack versus turbine building vents.*

NMPC Response:

The requested basis information was included in the discussion provided in the response to Item 5 above.

9. *Provide the historical results of leakage testing of the MSIVs.*

NMPC Response:

Main Steam Penetration X-2A

MSIV-01-01 (Inboard):

1989	17 SCFH	As-Found
1989	7.960 SCFH	As-Left
2/14/91	1.686 SCFH	As-Found
2/21/93	3.706 SCFH	As-Found
3/16/95	4.400 SCFH	As-Found
3/27/97	3.371 SCFH	As-Found

17 47 17
1.7 2.2 1.7
3.7 6.0 3.7
4.4 7.16 4.4
3.4 5.35 3.4
6

MSIV-01-03 (Outboard):

1989	47 SCFH	As-Found
1989	1.780 SCFH	As-Left
2/14/91	2.190 SCFH	As-Found
2/21/93	5.980 SCFH	As-Found
3/23/93	1.350 SCFH	As-Found
2/16/95	Un-quantified	As-Found
3/17/95	7.160 SCFH	As-Left
3/28/97	5.350 SCFH	As-Found

Main Steam Penetration X-2B

MSIV-01-02 (Inboard):

1989	11.8 SCFH	As-Found
1989	2.000 SCFH	As-Left
1/5/91	0.528 SCFH	As-Found
2/21/93	Un-quantified	As-Found
3/12/93	1.460 SCFH	As-Left
3/17/95	1.910 SCFH	As-Found
3/29/97	0.121 SCFH	As-Found

11.8 74 11.8
0.5 1.2 0.5
1.5 3.6 1.5
1.9 60 1.9
0.12 07 0.1
.9

MSIV-01-04 (Outboard):

1989	74.3 SCFH	As-Found
1989	1.150 SCFH	As-Left
1/5/91	3.580 SCFH	As-Found
3/9/93	60 SCFH	As-Found
4/1/93	3.530 SCFH	As-Left
3/12/95	2.470 SCFH	As-Found
3/25/97	Un-quantified	As-Found
3/25/97	0.083 SCFH	As-Left

10. *What is the fraction of the break flow associated with the LOCA that bypasses the suppression pool?*

NMPC Response:

In the previous LOCA analysis, no credit was taken for scrubbing in the suppression pool. Therefore, for the purpose of radiological analysis, 100% of the break flow was assumed to bypass the suppression pool. In the revised analysis, to be transmitted per Item 4 above, 2% of the break flow is assumed to bypass the suppression pool.

11. *While it is understood that the design basis of the reactor building emergency ventilation system is to remove one volume per day, it would be anticipated that one would take the free air volume of the reactor building, based upon that volume determine the appropriate fan flow rate. In the LOCA analysis which was provided the licensee determined the reactor building volume using fan capacity. That seems inappropriate because the fan velocity is dependent upon the fan curve and the fan may be over or under designed.*

NMPC response:

The dose assessment calculation for the LOCA that will be transmitted for your review per Item 4 above will use the calculated free volume of the reactor building.