LP-25

NIAGARA MOHAWK

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January 20, 1986
(NMP2L 0589)

Ms. Elinor G. Adensam, Director
BWR Project Directorate No. 3
U.S. Nuclear Regulatory Commission
7920 Norfolk Avenue
Washington, DC 20555

Dear Ms. Adensam:

Re: Nine Mile Point Unit 2
Docket No. 50-410

Attached is the Nine Mile Point Unit 2 response to the letter from W. Butler (NRC) to B. G. Hooten (NMPC), dated November 15, 1986 concerning conformance to Regulatory Guide 1.97.

This material provides the information necessary to close Confirmatory Item 10.

Very truly yours,

CVMangan

C. V. Mangan
Senior Vice President

TRL:ja
Attachment

xc: R. A. Gramm, NRC Resident Inspector
Project File (2)

8601240094
8 pp.

ATTACHMENT

1. The licensee should identify plant-specific Type A variables and verify that the instrumentation for them is Category I.

   Response

   Table 421.36-1 of the Nine Mile Point Unit 2 Final Safety Analysis Report will be revised to identify the following Type A variables.

   1. Containment hydrogen concentration (2) (same as variables C11a and b)

   2. Containment oxygen concentration (2) (same as variables C12a and b)

   3. Reactor vessel pressure (2) (same as variables B6a and b)

   4. Reactor vessel level (4) (same as variables B4a, b, c and d)

   5. Suppression pool water temperature (8) (same as variables D6a and b)

   6. Drywell atmosphere temperature (18) (same as variables D7a and b)

   7. Drywell atmosphere pressure (2) (same as variables D4a and b)

2. Neutron flux - The applicant should provide redundant Class 1E power sources for this instrumentation; the applicant should show that the source and intermediate ranges have sufficient overlap.

   Response

   The source of power (see Nine Mile Point Unit 2 Final Safety Analysis Report Figure 8.3-10) for Source Range Monitors (SRMs) and Intermediate Range Monitors (IRMs) originates from reliable normal dc sources. Normal supply originates from stub buses (2NJS-US6 and US5) to 24/48-V dc distribution panels (2BWS-PNL300A and 2BWS-PNL300B). A normal power source feeds two battery chargers per division that service the 24/48-V dc distribution panel(s) and maintain the charge on two 24-V dc batteries. The batteries are available to service the 24/48-V dc distribution panels on loss of normal power.

   The power source for Low Power Range Monitor (LPRM) groups and Average Power Range Monitor (APRM) channels is from the RPS/UPS channelized Divisions 1 through 4. This power is fed to RPS buses by means of a UPS which has normal, alternate and battery backup sources. Power sources are channelized RPS Divisions 1 through 4. The power distribution system for this instrumentation is described in detail in Section 8.3.1.1.3 of the Final Safety Analysis Report.

   It is Niagara Mohawk's determination that this design provides reliable power sources.

Although the power supplies are classified as nonsafety-related, this does not impair the ability of the neutron monitoring instrumentation to perform its required detection and trip functions. The trip function is configured to trip and initiate a scram on loss of power. The instrumentation is seismically and environmentally qualified, so the trip function is ensured on loss of power.

The operating ranges of the source range (SRM) and intermediate range (IRM) devices are as follows:

SRM = 1×10^3 to 1.5×10^9 nv
IRM = 1×10^8 to 1.5×10^{13} nv

The overlap of the ranges is 1×10^8 to 1.5×10^9 nv.

Additionally, there is a typographical error in Table 421.36-1. The lower end of the IRM range should be 4.0×10^{-5} percent power. This will be corrected in the next Final Safety Analysis Report amendment.

3. Reactor coolant system soluble boron concentration - The applicant should identify the range of the instrumentation being supplied for this variable.

   Response

   The range is 50 to 2,000 ppm boron in solution. This information will be incorporated in the next Final Safety Analysis Report update.

4. Coolant level in reactor - The applicant should identify the remainder of this instrumentation in accordance with Section 6.2 of NUREG-0737, Supplement No. 1, identify any deviations, and justify those deviations identified.

   Response

   Conformance to Regulatory Guide 1.97, Revision 3, is accomplished by the use of two transmitters per division, one fuel zone and one wide range. As noted in Section 3.3.3 of your November 15, 1985 letter, the wide range transmitters were omitted from the table and will be added in the next Final Safety Analysis Report amendment.

   The wide range transmitters (2ISC*LT9C and D) are calibrated to monitor the 375.70 in. level, which is inside the fuel zone transmitter range, to the 585.70 in. level which is 62.3 in. below the centerline of the main steam lines at 648 in.

   This range is considered to meet the intent of the regulatory guide which is to restore and maintain reactor pressure vessel water level to ensure adequate core cooling.

5. Drywell pressure - The applicant should provide independent Class 1E power sources for these instrument channels.

Response

The two instrument channels are powered from separate Class 1E sources. Table 421.36-1 is incorrect and will be corrected in a future Final Safety Analysis Report amendment.

6. Drywell sump level - The applicant should provide instrumentation for this variable.

Response

See Item 7.

7. Drywell drain sumps level - The applicant should provide instrumentation for this variable.

Response (Items 6 and 7)

The drywell sump level instruments provide indication of identified and unidentified leakage during normal operating conditions. The instrumentation, which is nonsafety grade (Category 3), is located on drain tanks in the secondary containment, outside of the drywell. Under accident conditions, these drain tanks are automatically isolated from the primary containment to prevent the escape of any post-accident reactor fluid from the drywell. In this situation, the drywell sump level indication is no longer meaningful and thus serves no post-accident safety function.

Other instrumentation is available to identify leakage into the drywell. This includes drywell pressure, drywell temperature, and containment radiation. These instruments meet the Category I requirements of Regulatory Guide 1.97.

8. Radiation level in circulating primary coolant - The applicant should supply the recommended instrumentation and the information required by Section 6.2 of NUREG-0737, Supplement No. 1, identify any deviations from the regulatory guide, and justify those deviations.

Response

This instrumentation is not provided at Nine Mile Point Unit 2.

Justification

The usefulness of information obtained by monitoring the radiation level in the circulating primary coolant, in terms of helping the operator in his efforts to prevent and mitigate accidents has not been substantiated. The particular planned operator action to be taken based on monitoring this variable is not specified in the current draft of the Emergency Procedures. The critical actions taken to prevent and mitigate a gross breach of fuel cladding are to shutdown the reactor and maintain water level. Monitoring primary coolant radioactivity has no influence on either of these actions. The purpose of this monitor falls in the cateogry of "information that the barriers to release of radioactive material are being challenged" and

"identification of degraded conditions and their magnitude, so the operator can take actions that are available to mitigate the consequences." Additional operator actions to mitigate the consequences of fuel barriers being challenged, other than those based on Type A and B variables, have not been identified.

Regulatory Guide 1.97 specifies measurement of the radioactivity of the circulating primary coolant as the key variable in monitoring fuel cladding status during isolation of the nuclear steam supply system (NSSS). The words "circulating primary coolant" are interpreted to mean coolant, or a representative sample of such coolant, that flows past the core. A basic criterion for a valid measurement of the specified variable is that the coolant being monitored is coolant that is in active contact with the fuel, i.e., flowing past the failed fuel. Monitoring the active coolant (or a sample thereof) is the dominant consideration. The post-accident sampling system (PASS) provides a representative sample which can be monitored.

The concern of Regulatory Guide 1.97 assumes a situation in which the NSSS is isolated and the reactor is shutdown. This assumption is justified because the monitors in the off-gas system and main steam tunnel provide reliable and accurate information on the status of fuel cladding when the plant is not isolated. Further, the PASS, once activated, provides an accurate status of coolant radioactivity and hence, cladding status. In the interim between NSSS isolation and operation of the PASS, monitoring of the primary containment radiation and hydrogen levels provides information on the status of the fuel cladding.

Present emergency procedures provide that once initial core damage is estimated using information obtained from the analysis of PASS samples, the estimate is confirmed using containment hydrogen analysis, containment high-range radiation monitoring, water level indications, and Sr, Ba, La, and Ru analyses. Therefore, no Type C Category I instrumentation is provided to measure the subject variable.

The Niagara Mohawk position agrees with the BWR Owners Group position on this variable.

9. Analysis of primary coolant - The applicant should identify the range of the instrumentation being supplied for this variable.

<u>Response</u>

The instrument range is 10^{-6} to 10^{1} Ci/gm and will be incorporated in the next Final Safety Analysis Report update.

10. Radiation exposure rate - The applicant should show that the ranges encompass the expected radiation levels in their locations.

<u>Response</u>

Access is not required in any area of secondary containment to service safety-related equipment in a post-accident situation. When accessibility is reestablished in the long term, it will be done by a combination of portable radiation survey instruments and post-accident sampling of the secondary containment atmosphere.

Area monitors provided in areas outside secondary containment where access may be required post-accident have ranges that envelop the dose rates expected in these areas at the time access is required.

11. Residual heat removal heat exchanger outlet temperature - Environmental qualification should be addressed in accordance with 10CFR50.49.

<u>Response</u>

The instrumentation for this variable is Category I and environmentally qualified to 10CFR50.49. It is in compliance with the requirement for Regulatory Guide 1.97, Category 2 instrumentation. Table 421.36-1 of the Final Safety Analysis Report is incorrect and will be corrected in a future amendment.

12. Cooling water temperature to engineered safety feature system components - The applicant should justify the deviation in range.

<u>Response</u>

The temperature range of 2SWP*TE31A and B complies with the intent of Nuclear Regulatory Commission Regulatory Guide 1.97, Revision 3. The intent of the regulatory guide is to ensure that instrument ranges are selected so that the instrument will always be on scale. 2SWP*TE31A and B are located on the service water supply header and are used to monitor service water supply temperature. The temperature range of service water instrument is based upon the range for Lake Ontario, which normally varies from 38°F to 77°F and which is well within the instrument range of 32°F to 130°F.

13. Secondary containment area radiation - The applicant should supply the information required by Section 6.2 of NUREG-0737, Supplement No. 1, identify any deviations, and justify those deviations; environmental qualification should be addressed in accordance with 10CFR50.49.

<u>Response</u>

Environmentally qualified area monitors with ranges of 10^{-1} to 10^4 R/hr are not provided in secondary containment at Nine Mile Point Unit 2.

<u>Justification</u>

The use of local area radiation monitors to detect breach or leakage through primary containment penetrations is inappropriate. In general, radiation levels in the secondary containment will be largely a function of radioactivity in primary containment and in the fluids flowing in emergency core cooling system (ECCS) piping. Localized hot spots due to piping sources and primary containment penetrations and hatches will provide ambiguous indications. Breach of primary containment will be detected by the reactor building exhaust gaseous effluent monitor prior to the reactor building isolation and the noble gas channel of the main stack gaseous effluent monitor following the isolation of the reactor building. Therefore, these area monitors are not necessary and are not implemented at Nine Mile Point Unit 2.

In the long term, accessibility to the secondary containment will be reestablished using a combination of portable radiation survey instruments and post-accident sampling of the secondary containment atmosphere.

14. Noble gas, radwaste vent - The applicant should either provide the recommended range or justify the use of the lesser range.

Response

See Item 15.

15. Noble gas, common plant vent - The applicant should either provide the recommended range or justify the use of the lesser range.

Response (Items 14 and 15)

The noble gas channels of the gaseous effluent monitors for the radwaste/reactor building vent and the main stack release points have design ranges of 10^{-6} to 10^{4} uCi/cc, which meet Regulatory Guide 1.97 requirements.

16. Plant and environs radiation - The applicant should identify the ranges of this instrumentation and show that the ranges are adequate.

Response

Two types of portable radiation detection and instrumentation are provided to monitor the plant and environs. An ion chamber detector with a range of 10^{-3} to 50 R/hr is used for low-level gamma and beta radiation monitoring. A Geiger-Muller Teletector type detector with a range of 10^{-2} to 10^{3} R/hr is used for high level gamma radiation monitoring. With a combined range of 10^{-3} to 10^{3} R/hr, these instruments have adequate range to envelop the dose rates expected outside the plant buildings after an accident.

17. Accident sampling (primary coolant, containment air and sump) - The applicant should provide the information required by Section 6.2 of NUREG-0737, Supplement No. 1, identify any deviations from the regulatory guide, and justify those deviations.

Response

1. Instrument range - Analysis range is given in Table II.B.3-2 (See Section 1.10, Table II.B.3-2 of Nine Mile Point Unit 2 Final Safety Analysis Report). The ranges meet or exceed requirements of Regulatory Guide 1.97, within instrument limitations, with the exception of the dissolved gas sample analysis. The ranges given for dissolved gas analysis were approved by the Nuclear Regulatory Commission in a letter to General Electric (letter from W. Johnston [Nuclear Regulatory Commission] to G. Sherwood [General Electric] dated July 17, 1984).

2, 3, 4, 6. Environmental qualification, seismic qualification, quality assurance, and power supply - These have been addressed in Table 421.36-1 and meet the requirements of Regulatory Guide 1.97.

5. Redundancy and sensor locations - This is not applicable, since analysis is done in a chemistry laboratory on grab samples. Regulatory Guide 1.97 has no specific provision.

7. Display location - This is not applicable, since analysis is done in a chemistry laboratory, the display is on each individual instrument. This meets the requirements of Regulatory Guide 1.97.

18. Primary containment isolation valve position - The applicant should provide justification for the exemption of instrumentation for the traversing incore probe system isolation valves.

Response

The traversing incore probe (TIP) system isolation valves consist of ball valves, operated when the probe is out of the guide tube, and shear valves manually operated if the probe is in the guide tube.

The TIPs are normally withdrawn and the ball valves are closed. If an event occurs while the TIP is inserted into the core and the TIP should fail to retract, the shear valve can be operated manually to provide the necessary containment isolation.

These valves are classified nonessential and are provided with non-Class 1E automatic isolation signals and power and as such, cannot meet Regulatory Guide 1.97, Category 1 or 2 requirements. (For further explanation, refer to the Nine Mile Point Unit 2 Final Safety Analysis Report Section 1.10, TMI action item II.E.4.2 concerning the Containment Isolation Dependability.)

Additionally, any leakage through this line has been incorporated in the radiological LOCA analysis of Chapter 15.6.5 of the Nine Mile Point Unit 2 Final Safety Analysis Report.

19. Please verify that Category I instrumentation is or will be provided for neutron flux instrumentation (from W. R. Butler letter of October 15, 1985). (Table 421.36-1 of Final Safety Analysis Report commits to environmentally and seismically qualified equipment.)

Response

Neutron flux measurement devices located in harsh environment areas are environmentally and seismically qualified for the anticipated environments. The environmental qualification under harsh environment conditions is for a limited time, but the time is sufficient to perform the detection, mitigation, and monitoring functions required of the instrumentation. Instrumentation located in mild environment areas is seismically qualified.