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SUBJECT: Forwards response to NRC 810831 request for info re NUREG-0803, "Generic Safety Evaluaton Rept re Integrity of BWR Scram Sys Piping."

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December 31, 1981

Mr. Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation
United States Nuclear Regulatory Commission
Washington, D. C. 20555

Re: Nine Mile Point Unit 1
Docket No. 50-220

Dear Mr. Eisenhut:

Your letter dated August 31, 1981 requested licensees to provide information regarding NUREG 0803, "Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping." The attachment to this letter provides that information.

Very truly yours,

NIAGARA MOHAWK POWER CORPORATION

D P Dise

D. P. Dise
Vice President Engineering

BDW:ja
Attachment

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NINE MILE POINT UNIT 1
RESPONSE TO NRC GUIDANCE
FOR RESOLVING SAFETY CONCERNS
ASSOCIATED WITH PIPE BREAKS IN THE BWR SCRAM SYSTEM

Section 5 of NUREG 0803, "Generic Safety Evaluation Report Regarding the Integrity of Boiling Water Reactor Scram System Piping" contains the NRC staff's generic conclusions on this issue. The report identifies three major areas of concern:

1. The integrity of the scram system piping,
2. The capability for mitigating a pipe break in the scram system, and
3. The environmental qualification of equipment which would be used to identify and mitigate a pipe break in the scram system.

Information which addresses these concerns is provided herein for Nine Mile Point Unit 1. Specifically, the NRC guidance for resolving these concerns, as summarized in Table 5.1 of NUREG 0803, is addressed.

1. <u>Piping Integrity</u>

   Section 5.1 and Table 5.1 of NUREG 0803 provide the following guidance to ensure the integrity of the scram discharge volume piping.

   a. Periodic in-service inspection and surveillance for the scram discharge volume system,

   b. Seismic design verification,

   c. Hydraulic control unit - scram discharge volume equipment procedures review, and

   d. As-built inspection of scram discharge volume piping and supports.

   Niagara Mohawk is committed to implementation of this guidance, as discussed below.

   The scram discharge system piping will be included in the Nine Mile Point Unit 1 in-service inspection program in accordance with the requirements for Class 2 piping in Section XI of the ASME Boiler and Pressure Vessel code. Inspection of 100 percent of the required welds is anticipated by the end of the scheduled 1985 maintenance and refueling outage.

   Recent modifications to the scram discharge volume system included the design and installation of a support system capable of accommodating seismic loads as well as other loads anticipated during scram. All design and analyses were performed in accordance with subsection NC of Section III of the ASME Boiler and Pressure Vessel code, 1977 edition. An analysis of the as-built configuration of the scram discharge volume piping is currently underway. Completion of this analysis is anticipated by March 1982.

   Hydraulic control unit-scram discharge volume equipment maintenance and surveillance procedures have been reviewed to ensure that pressure boundary integrity is maintained during maintenance or surveillance testing. The review identified the need to modify Surveillance Procedure N1-ISP-RD-08, "High Level Scram Discharge Volume Instrument Channel Calibration." The existing procedure does not require isolation of the equipment being calibrated. This procedure will be revised by March 1982 to require equipment isolation prior to calibration.

2. Mitigation Capability

Section 5.2 of NUREG 0803 states that successful mitigation of a pipe break in the scram discharge piping is primarily dependent on equipment qualification and a prompt blowdown of the reactor vessel. NUREG 0803 suggests that the BWR Owners Group emergency procedures guidelines should be revised to "lead the control room operator into a controlled blowdown when plant conditions indicate a reactor system leak outside the drywell."

The BWR Owners Group has discussed the guidance of NUREG 0803 regarding modification of the emergency procedure guidelines. It acknowledges the benefits of treating the subject generically. The BWR Owners Group is in the process of completing an extension of the guidelines to include steps for reactivity control, and certain other modifications to the guidelines which have been discussed with your staff. It is Niagara Mohawk's judgment that completion of these modifications outweighs, in immediate importance, the NUREG 0803 guidance for other guideline modifications. Therefore, after current activities on the guidelines are substantially complete, Niagara Mohawk will support a preliminary study by the BWR Owners Group to determine the best approach to fulfilling the intent of the guidance provided in NUREG 0803. It is not clear that the best approach will involve modification of the guidelines. Upon completion of the study currently expected near the end of the first quarter of 1982, the Owners Group will determine whether to authorize specific actions to modify the emergency procedure guidelines. Niagara Mohawk will advise you of the results of that decision in the Owners Group's plan at that time.

Section 5.2 of NUREG 0803 also states that in order to ensure access to the reactor building, all Boiling Water Reactors should implement the Standard Technical Specification limits for coolant activities. Demonstrating a low probability of operating in excess of Standard Technical Specification limits precludes the need for implementing them.

A review of coolant isotopic analyses for 1980 and 1981 indicates that Dose Equivalent I-131 concentrations were far below Standard Technical Specification limits. It is Niagara Mohawk's opinion that the results of this review, in conjunction with the lack of fuel failures during the last several years, provides sufficient assurance that coolant activity levels will not exceed standard Technical Specification limits. Therefore, the current Nine Mile Point Unit 1 technical specification for coolant activity levels will not be revised.

3. Environmental Qualification

Section 5.3 of NUREG 0803 requests licensees to:

a. Identify the equipment that would be used to detect a break and/or leak in the scram discharge volume system and include the qualification of this equipment in the NRC's ongoing Environmental Qualification program to show that it would perform the identification function.

b. Identify the equipment needed to mitigate an unisolable break in the scram discharge volume system and include the qualification of this equipment in the NRC's ongoing Environmental Qualification program to show that it would perform the mitigation function, paying particular attention to the guidance summarized in Table 5.1

c. For any equipment required for identification and/or mitigation that is not qualified for service at 212°F and 100% humidity, provide a schedule for defining the plant-specific scram discharge volume break environment and a commitment to qualify the equipment in accordance with the NRC's ongoing Environmental Qualification program.

A myriad of equipment is available to provide the operator with indication of a pipe break in the reactor building. Principal indication of a break in the scram discharge volume piping would come from high area radiation signals, high radiation from the continuous air monitors, and reactor building floor drain sump level alarms, in conjunction with personnel observation of leakage.

The radiation monitor located in the control rod drive/hydraulic control unit vicinity would probably provide the first indication of a break. Procedures require a survey to be taken upon receipt of a high area radiation signal. The conservative estimates provided in Table 4.2 of NUREG 0803 for Standard Technical Specification coolant activity limits indicate that dose rates in the vicinity of the leak would not prohibit access within the first few hours of the event. As noted in Section 2, coolant activity levels are normally below Standard Technical Specification limits. Therefore, dose rates would not prevent entry to the break vicinity. The steam environment may limit the ability of the team to enter the immediate control rod drive/hydraulic control unit area. However, the configuration of the Nine Mile Point Unit 1 plant would still enable the team to observe the scram discharge volume piping area from a distance. Thus, the survey team would provide personnel observation and positive verification of a break in the scram discharge volume piping.

Niagara Mohawk will evaluate installation of equipment qualified for operation in a 212°F, 100% humidity environment which would provide unambiguous indication of leakage from the scram discharge volume piping. Therefore, the equipment identified above will not be included in the Inspection and Enforcement 79-01B qualification effort at this time. However, if the evaluation indicates that installation of additional break detection equipment is not feasible or is undesireable, then the area radiation monitor, the continuous air monitor, and the reactor building floor drain sump level equipment will be qualified for operation in the environment resulting from a break in the scram discharge volume piping.

Equipment which might be used to mitigate a break in the scram discharge volume piping includes the feedwater/HPCI system, the control rod drive pumps, the automatic depressurization system, the emergency condensers, and the core spray system.

Immediately following a break, the vessel level would be maintained automatically by the feedwater/HPCI system. Automatic control of vessel level requires the feedwater/HPCI level sensing and signal transmitting equipment to be operable. This equipment is located in the reactor building and will be qualified, as a minimum, for operation in the environment which would result from a pipe break in the scram discharge volume piping. Manual operation of the feedwater system is independent of reactor building environment.

The automatic depressurization system could be used to depressurize the reactor and hence reduce leakage resulting from a break in the scram discharge volume piping. Electrical components of the automatic depressurization system which are located in the reactor building are included in the Inspection and Enforcement 79-01B program and will be qualified, as a minimum, for operation in the environment which would exist following a break in the scram discharge volume piping.

The emergency condensers provide a means of quickly reducing the temperature and pressure of the reactor vessel without resorting to the use of the automatic depressurization system. Use of the emergency condensers would keep reactor coolant within the primary system, making it easier to control reactor vessel water level. Test data indicate that the Emergency Condensers have a combined heat removal rate of approximately 650×10^6 BTU/hr with the reactor at 930 psia. Based on this data, it is estimated that the Emergency Condensers could be used to reduce the reactor from operating pressure to less than 50 psia in approximately 90 minutes. As indicated in NEDO 24342, at this reduced pressure the leakage rate would not prohibit access to the manual hydraulic control unit isolation valves. Since the emergency condensers provide an alternative to the automatic depressurization system, they are not considered the primary means for reducing reactor temperature and pressure. Therefore, the electrical components of the emergency condenser system are not included in the Inspection and Enforcement Bulletin 79-01B qualification effort.

The control rod drive pumps can be used to provide makeup to the vessel in the event offsite power and hence feedwater is unavailable. At reduced reactor pressure, such makeup would be sufficient to maintain reactor water level. In addition, the control rod drive pumps can be used to provide cooling water to the control rod drive seals and would significantly reduce the temperature and activity of the water discharged from a scram discharge volume pipe break. The reduced temperature of the discharge water would slow down the degradation of the reactor building environment and allow quicker, if not immediate, access to the manual isolation valves. Environmental Qualification of the control rod drive pump motors is presently ongoing and is included in the Inspection and Enforcement Bulletin 79-01B qualification program. The pump motors will be qualified, as a minimum, for operation in the environment which would result from a scram discharge volume pipe break.

The core spray system provides a means of injecting water into the vessel in the event a pipe break in the scram discharge volume system occurs concurrently with a loss of offsite power. The use of the core spray system in conjunction with either the emergency condensers or the automatic depressurization system can provide sufficient makeup to the vessel to offset leakage resulting from a scram discharge volume pipe break. Electrical equipment associated with operation of the core spray system is currently included in the qualification efforts of Inspection and Enforcement Bulletin 79-01B and will be qualified, as a minimum, for operation in the reactor builing environment resulting from a scram discharge volume pipe break.

Section 4.3.3 of NUREG 0803 expresses concern that the ECCS pump motors could be subject to wetdown by 212°F water. The location of the core spray pumps at Nine Mile Point Unit 1 precludes significant wetdown of the pump motors from occurring. Minor splashing might occur, but it would not be expected to interfere with pump operation. However, the core spray pump motors will be qualified for wetdown by 212° water.

Implementation of the guidance discussed in Section 1, Piping Integrity, will significantly reduce the probability of occurrence of a scram discharge volume pipe break. Furthermore, the information provided in Sections 2 and 3 demonstrates that such a pipe break would be detected and mitigated using existing equipment. This information demonstrates that continued operation of Nine Mile Point Unit 1 does not present an undue safety hazard to the public.

[illegible]