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AUTH.NAME AUTHOR AFFILIATION
DISE,D.P. Niagara Mohawk Power Corp.
RECIP.NAME RECIPIENT AFFILIATION
EISENHUT,D.G. Division of Licensing

SUBJECT: Forwards evaluation of GE 800221 generic response to Michelson concerns re facility. Evaluation responds to NRC 800507 request re NUREG-0660, Item 2.K.3.46.

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NIAGARA MOHAWK

NIAGARA MOHAWK POWER CORPORATION/300 ERIE BOULEVARD WEST, SYRACUSE, N.Y. 13202/TELEPHONE (315) 474-1511

August 1, 1980

Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Eisenhut:

Re: Nine Mile Point Unit 1
Docket No. 50-220
DPR-63

Attached are the results of Niagara Mohawk's evaluation of General Electric's February 21, 1980 generic response to the Michelson's concerns for Nine Mile Point Unit 1. This information was requested in your letter of May 7, 1980 in response to NUREG-0660, Item 2.K.3.46 Response to List of Concerns from ACRS Consultant.

We are in agreement with General Electric's generic response for BWR boiling water reactors, except as noted in the attached evaluation.

Very truly yours,

NIAGARA MOHAWK POWER CORPORATION

Donald P. Dise
Vice President - Engineering

PEF:ja
Attachment

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NIAGARA MOHAWK'S ASSESSMENT OF THE APPLICABILITY OF
GENERAL ELECTRIC'S RESPONSE TO THE
MICHELSON'S CONCERNS TO NINE MILE POINT UNIT 1

Question 1

Pressurizer level is an incorrect measure of primary coolant inventory.

General Electric's Response

BWRs do not have pressurizers. BWRs measure primary coolant inventory directly using differential pressure sensors attached to the reactor vessel. Thus, this concern does not apply to BWRs.

Niagara Mohawk's Assessment

Niagara Mohawk agrees with General Electric's response.

Question 2

The isolation of small breaks (e.g., letdown line; PORV) not addressed or analyzed.

General Electric's Response

Automatic isolation only occurs for breaks outside the containment. Such breaks are addressed in Section 3.1.1.1.2 of NEDO-24708. It was shown that if the high pressure systems are available no operator actions are required. If it is assumed that all high pressure systems to inject and maintain vessel water level. Analyses submitted for demonstration of adequate core cooling* show that the operator has sufficient information and time to perform these manual actions. The necessary manual actions have been included in the operator guidelines for small break accidents.

Niagara Mohawk's Assessment

Niagara Mohawk agrees with General Electric's response. However, it should be noted that this discussion applies to breaks outside containment.

Question 3

Pressure boundary damage due to loadings from a) bubble collapse in subcooled liquid and 2) injection of ECC water in steam-filled pipes.

General Electric's Response

The BWR has no geometry equivalent to that identified in Michelson's report on B&W reactors relative to bubble collapse (steam bubbling upward through the pressurizer surge line and pressurizer). Thus the first concern is not applicable to BWRs.

* NEDO-24708 Section 3.5.2.1, which was submitted in prepublication form to D.F. Ross by letter from R.H. Buchholz on November 30, 1979.

Question 3 (cont'd)

General Electric's Response (cont'd)

ECC injection in the BWR at high pressure is either directly into the reactor vessel (BWR/5-6 HPCS, HPCI on some BWR/4) or into the feedwater lines (FWCI, HPCI on most BWR/3-4). The feedwater lines are normally filled with relatively cold liquid (420°F or less). ECCS injection in the BWR low pressure is either directly into the reactor vessel (LPCS, BWR/5-6 LPCI) or into the recirculation pump discharge line (BWR/3,4 LPCI) near the automatically-closed recirculation pump discharge valve. Thus the second concern is not applicable to BWRs.

Niagara Mohawk's Assessment

Niagara Mohawk agrees with General Electric's response that the BWR has no geometry equivalent to that identified in Michelson's report on B&W reactors relative to bubble collapse (steam bubbling upward through the pressurized surge line and pressurizer). However, the injection of ECC water at low pressure at Nine Mile Point Unit 1 is directly into the reactor vessel via the core spray piping. The piping between the reactor vessel nozzle and the core spray sparger could potentially contain steam. The adequacy of this sytem to withstand injection loads with contained steam will be investigated as part of our ongoing core spray sparger replacement study.

Question 4

In determining need for steam generators to remove decay heat, consider that break flow enthalpy is not core exit enthalpy.

General Electric's Response

BWRs do not use steam generators to remove decay heat, so this concern does not apply to BWRs. The GE modelling of break flow is discussed in NEDO-20566.

Niagara Mohawk's Assessment

Niagara Mohawk agrees with General Electric's response.

Question 5

Are sources of auxiliary feedwater adequate in the event of a delay in cooldown subsequent to a small LOCA?

General Electric's Response

BWRs do not need feedwater to remove heat from the reactor following a LOCA, whether the subsequent cooldown is delayed or not. Therefore, this concern is not applicable to BWRs. BWRs have a closed cooling system in which vessel water flows out the postulated break to the suppression pool. The suppression pool is cooled and water is pumped back to the vessel with ECCS pumps.

Niagara Mohawk's Assessment

Niagara Mohawk agrees with General Electric's response.

Question 6

Is the recirculation mode of operation of the HPCI pumps at high pressure an established design requirement?

General Electric's Response

The high-pressure injection systems utilized in the BWRs are the Reactor Core Isolation Cooling (RCIC), High Pressure Coolant Injection (HPCI) and High Pressure Core Spray (HPCS).

BWR/1 and 2 units do not have special-purpose high-pressure safety injection systems, such as HPCI and RCIC*. Some BWR's 3 and all BWR's 4, 5 and 6 are provided with RCIC systems. Some BWR's 3 and all BWR's 4 are provided with HPCI systems. BWR's 5 and 6 are provided with HPCS systems. The RCIC and HPCI are steam turbine driven variable speed systems. The HPCS is an electrically driven constant speed system.

The RCIC, HPCI and HPCS systems normally take suction from the condensate storage tank and have an alternate suction source from the suppression pool. A recirculation mode of operation of these systems is established when the system suction is from the suppression pool. Following a LOCA when system suction is from the suppression pool, water injected into the reactor is discharged through the break and flows back to the suppression pool forming a closed recirculation loop.

Other recirculation modes include test modes (e.g., suction from and discharge to the condensate storage tank) and system operation on low flow bypass with discharge to the suppression pool.

All of these modes are established design requirements.

Niagara Mohawk's Assessment

Niagara Mohawk agrees with General Electric's response. Nine Mile Point Unit 1, a BWR 2, does not have a steam turbine driven HPCI or RCIC or a HPCS. Nine Mile Point Unit 1 does have a HPCI system which utilizes the feedwater system. This system does not have a recirculation mode of operation. Therefore, this concern does not apply to Nine Mile Point Unit 1.

Question 7

Are the HPCI pumps and RHR pumps run simultaneously? Do they share common piping?/suction? If so, is the system properly designed to accomodate this mode of operation (i.e., are any NPSH requirements violated, etc....?)

General Electric's Response

As noted in response to Question No. 6, BWR/1 and 2 units do not have special-purpose HPCI or RCIC systems.

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* One unit is installing a high-pressure injection system -- it is not yet in service.

Question 7 (cont'd)

General Electric's Response (cont'd)

For BWR/3-6 the high-pressure injection systems (RCIC/HPCI/HPCS) do not share any common suction piping with the RHR, Low Pressure Coolant Injection (LPCI), or low-pressure core spray systems, and they can operate simultaneously with these low pressure systems.

The RCIC/HPCI and RCIC/HPCS systems, on some BWR's, share a common suction line from the condensate storage tank. Many of the BWR LPCI pumps (LPCI on most BWR's is a subsystem of the RHR) and low pressure core spary (CS) pumps share common suction piping. The RHR shutdown cooling operating mode does not share any common suction piping with the RCIC, HPCI, HPCS, CS/LPCS systems. It is an established design requirement to size the suction piping, including shared piping, such that adequate NPSH is available to RCIC, HPCI, HPCS, RHR/LPCI, and CS pumps for all simultaneous operating modes of these systems.

Pre-operational and/or startup tests are conducted that demonstrate that the requirement is met.

Niagara Mohawk's Assessment

At Nine Mile Point Unit 1 the HPCI pumps (which consist of the condensate pumps, the feedwater booster pumps and the motor driven feedwater pumps) do not take suction from the same water source as the emergency core cooling or heat removal systems. The low pressure core spray system which takes suction from the suppression pool does not share a common piping or suction with the containment spray system which is used for containment and suppression pool cooling.

Question 8

Mechanical effects of slug flow on steam generator tubes needs to be addressed (transitioning from solid natural circulation to reflux boiling and back to solid natural circulation may cause slug flow in the hot leg pipes).

General Electric's Response

BWRs do not have steam generators so this concern does not apply to BWRs. BWR post-LOCA cooling modes are addressed in NEDO-24708.

Niagara Mohawk's Assessment

Niagara Mohawk agrees with General Electric's response.

Question 9

Is there minimum flow protection for the HPCI pumps during the recirculation mode of operation?

Question 9 (cont'd)

General Electric's Response

As noted in response to Question No. 6, BWR/1 and 2 units do not have special purpose HPCS or RCIC systems.

For BWR/3-6, the RCIC, HPCI, HPCS, RHR, and CS/LPCS pumps all contain valves, piping, and automatic logic that bypasses flow to the suppression pool as required to provide minimum flow protection for all design basis operating modes of the systems.

Niagara Mohawk's Assessment

The Nine Mile Point Unit 1 HPCI system does not have a recirculation mode of operation.

Question 10

The effect of accumulators dumping during small break LOCAs is not taken into account.

General Electric's Response

BWRs do not use accumulators to mitigate LOCAs. Therefore, this concern does not apply to BWRs.

Niagara Mohawk's Assessment

Niagara Mohawk agrees with General Electric's response.

Question 11

What is the impact of continued running of the RC pumps during a small LOCA?

General Electric's Response

The impact of continued running of the recirculation pumps has been addressed in Sections 3.3.2.2, 3.3.2.3, and Section 3.5.2.1.5.1 of NEDO-24708*. The conclusions were that continued running of the recirculation pumps results in little change in the time available for operator actions and does not significantly change the overall system response.

Niagara Mohawk's Assessment

Niagara Mohawk agrees with General Electric's response.

* The latter section was transmitted in prepublication form to D.F. Ross by letter from R.H. Buchholz on November 30, 1979.

Question 12

During a small break LOCA in which offsite power is lost, the possibility and impact of pump seal damage and leakage has not been evaluated or analyzed.

General Electric's Response

The RCIC, HPCI, HPCS, RHR, CS/LPCS pumps are provided with mechanical seals. These seals are cooled by the pump primary process water. No external cooling from auxiliary support systems, such as site service water or room air coolers, is required for pump seals. These types of seals have demonstrated (in nuclear and other applications) their capability to operate for extended periods of time at temperatures in excess of those expected following a LOCA.

Should seal failure occur it can be detected by room sump high level alarms. The RCIC, HPCI, HPCS, LPCS and CS and RHR individual pumps are arranged, and motor operated valves provided, so that a pump with a failed seal can be shutdown and isolated without affecting the proper operation of the other redundant pumps/systems.

Considering the low probability of seal failure during a LOCA, the fact that a pump with a failed seal can be isolated without affecting other redundant equipment, and the substantial redundancy provided in the BWR emergency cooling systems, pump seal failure is not considered a significant concern.

Niagara Mohawk's Assessment

Niagara Mohawk agrees with General Electric's response for the Nine Mile Point Unit 1 Core Spray and Containment Spray pump seals. Regarding the Emergency Condenser System, which could also be utilized during a small break LOCA with a loss of offsite power, this item is not applicable. The Emergency Condenser System is gravity fed and has no pumps.

Question 13

During transitioning from solid natural circulation to reflux boiling and back again, the vessel level will be unknown to the operators, and emergency procedures and operator training may be inadequate. This needs to be addressed and evaluated.

General Electric's Response

There is no similar transition in the BWR case. In addition, the BWR has water level measurement within the vessel and the indication of the water level is incorporated into the operator guidelines. Consequently, this concern does not apply to BWRs.

Niagara Mohawk's Assessment

Niagara Mohawk agrees with General Electric's response.

Question 14

The effect on non-condensible gas accumulation in the steam generators and its possible disruption of decay heat removal by natural circulation needs to be addressed.

General Electric's Response

The effect of non-condensible gas accumulation is addressed in Section 3.3.1.8.2 of NEDO-24708. For a BWR, vapor is present in the core during both normal operation and natural circulation conditions. Non-condensibles may change the composition of the vapor but would have an insignificant effect on the natural or forced circulation itself, since the non-condensibles would rise with the steam to the top of the vessel after leaving the steam separators.

Niagara Mohawk's Assessment

Niagara Mohawk will address this concern by either demonstrating by analysis the ability of the Nine Mile Point Unit 1 Emergency Condensers to adequately cool the core with non-condensibles or modify the emergency condenser tube side vent to vent the condensers to the torus under accident conditions.

Question 15

Delayed cooldown following a small break LOCA could raise the containment pressure and activate the containment spray system. Impact and consequences need addressing.

General Electric's Response

A. Mark I and II Containments:

Except for a few early plants, most plants with Mark I and II containments do not have an automatically initiated drywell or wetwell spray. Only one of the newer plants (not yet operating) has an automatic wetwell spray.

It is very unlikely that the operator would manually initiate drywell sprays even given that a LOCA has occurred. Some non-essential equipment in the drywell (e.g., recirculation pumps) could be adversely affected by drywell has been qualified for the steam and temperature environment that would exist following a LOCA.

There is no equipment in the wetwell that is adversely affected by wetwell sprays.

B. Mark III Containments:

There is no drywell spray in a Mark III containment.

There is an automatic spray system in the wetwell (containment). All essential equipment has been qualified for this condition.

Question 15 (cont'd)

Niagara Mohawk's Assessment

Nine Mile Point Unit 1 design includes automatic initiation of the Containment Spray System. This sytem will cool the suppression pool water and spray the drywell and wetwell all at the same time. The affect of the activation of the Containment Spray System during a small break LOCA are the same for a large break LOCA. This concern is not as significant in a BWR as a PWR because most essential equipment is located outside of primary containment. Although some non-essential equipment in the drywell (e.g. recirculation pump motors) could be adversely affected by the sprays all essential equipment has been qualified for the effects of the Containment Spray System. There is no equipment in the suppression pools which could be affected by the Containment Spray System.

Question 16*

This concern relates to the possibility that an operator may be inclined and perhaps even trained to isolate, where possible, a pipe break LOCA without realizing that it might be an unsafe action leading to high pressure, and short-term core bakeout. For example, if a BWR should experience a LOCA from a pressure boundary failure somewhere between the pump suction and discharge valve for either reactor recirculation pump, it would be possible for the operator to close these valves following the reactor blowdown to low pressure and thereby isolate the break, stop the blowdown, and repressurize the reactor coolant system. Before such isolation should be permitted, it is first necessary to show by an appropriate analysis that the high pressure ECCS is adequate to reflood the uncovered core without assistance from the low pressure ECCS which can no longer deliver flow because of repressurization. Otherwise, such isolation action should be explicitly forbidden in the emergency operating instructions.

General Electric's Response

If a BWR should experience a LOCA from a pressure boundary failure somewhere between the recirculation pump suction and discharge valves, it is possible for the operator to close these valves following the reactor blowdown to low pressure and thereby isolate the break. In Reference 1, the NRC concluded based on information provided by GE that recirculation break isolation is not a problem.

In order for the reactor vessel to repressurize following isolation of a recirculation line break, the isolation would have to occur before initiation of ADS due to a high drywell pressure in concurrence with low water level 1 condition. Isolation of a recirculation line break prior to obtaining a high drywell pressure signal might occur for very small breaks (area $<<0.01$ ft^2) which may require several hundred seconds following the break to reach the high drywell pressure set point. In this case it has been shown in Reference 3 that the high pressure systems (RCIC, HPCI/HPCS, feedwater, and CRD) are sufficient to maintain the water level above the top of the core.

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* Excerpt from a memo, C. Michelson to D. Okrent, "Possible Incorrect Operator Action Such as Pipe Break Isolation", June 4, 1979.

Question 16 (cont'd)

General Electric's Response (cont'd)

If isolation of the recirculation break were to occur prior to reaching level 1 but after the high drywell pressure signal, the vessel would pressurize to the SRV set point following isolation of the main steam lines and then oscillate as the SRVs cycle open and closed. If no high pressure systems were available, the loss of mass out the SRVs would cause the level to continue dropping and result in automatic ADS actuation shortly after reacing level 1. This would depressurize the vessel and allow the low pressure systems to begin injecting. This capability was demonstrated in analyses presented in Reference 2. The small-break operator guidelines in NEDO-24708, in addition, explicitly provide for manual depressurization in the event of low reactor water level with high pressure systems unable to maintain level for any reason.

In summary, in order to repressurize the vessel following recirculation break isolation, the isolation would have to occur prior to ADS blowdown. For these cases, high pressure systems would maintain inventory and it has been shown that no adverse consequences result from isolation of a break in the recirculation line.

Niagara Mohawk's Assessment

Niagara Mohawk agrees with General Electric's response with the following additional information. The Nine Mile Point Unit 1 feedwater HPCI system is capable of preventing core uncover for breaks up to approximately 1.0 ft^2. This is within General Electric's analysis which indicated breaks $<<0.01$ ft^2 would cause repressurization of the reactor coolant system if a broken recirculation line is isolated. If the feedwater HPCI is unavailable upon repressurization of the reactor coolant system, the system can be depressurized by manual activation of the ADS. This would reduce the pressure sufficient to allow injection of the water by the low pressure Core Spray System.

1. Letter, D.C. Eisenhut to R.L. Gridley, "Potential for Break Isolation and Resulting GE-Recommended BWR/3 ECCS Modifications", June 14, 1978.

2. "Additional Information Required for NRC Staff Generic Report on Boiling Water Reactors", NEDO-24708, August 1979.

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2.	Perry 1/2	50-440, 50-441
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12.	Nine Mile Point 2	50-410
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15.	Limerick 1/2	50-352, 50-353
16.	Hope Creek 1/2	50-354, 50-355
17.	Marble Hill 1/2	50-546, 50-547
18.	Seabrook 1/2	50-443, 50-444
19.	Sterling 1	50-485
20.	Hartsville 1/2/3/4	50-518, 50-519, 50-520, 50-521
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27.	Catawba 1/2	50-413, 50-414

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1.	Pilgraim 2	50-471
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3.	Perkins 1/2/3	50-488, 50-489, 50-490
4.	Allens Creek 1	50-466
5.	Montague 1/2	50-496, 50-497
6.	New Haven 1/2	50-596, 50-597
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10.	Black Fox 1/2	50-556, 50-557
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12.	Davis Besse 2/3	50-500, 50-501
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15.	FNP	50-437

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