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ZWOLINSKI, J. A. BWR Project Directorate 1

SUBJECT: Forwards response to NRC 860508 safety evaluation re SPDS. Addl info re Sections 2.2 & 2.4 & electrical & electronic isolators described in Section 2.5 provided. Plant procedures revised to require display of SPDS.

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NIAGARA MOHAWK

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

September 29, 1986
NMP1L 0099

Director of Nuclear Reactor Regulation
Attention: Mr. John A. Zwolinski, Project Director
BWR Project Directorate Number 1
Division of BWR Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

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

Dear Mr. Zwolinski:

Enclosed is our response to the Safety Evaluation of the Nine Mile Point Unit 1 Safety Parameter Display System transmitted to us by your letter dated May 8, 1986. This response provides the additional information requested in Sections 2.2 and 2.4 of the Safety Evaluation. We have also provided additional information regarding electrical and electronic isolators described in Section 2.5.

Very truly yours,

NIAGARA MOHAWK POWER CORPORATION

CVMangan

C. V. Mangan
Senior Vice President

KBT:svm
Enclosure
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Response to NRC Safety Evaluation of

Nine Mile Point Unit 1

Safety Parameter Display System (SPDS)

2.2 SPDS Validation

Insufficient information is provided to evaluate the adequacy of the simulated input used in validation testing. Specifically, the transient and accident sequence test cases used for performance tests of the SPDS should be provided and justified. Reference to the transients identified in the report "Simulator Evaluation of the Boiling Water Reactor Owners Group (BWROG) Graphic Display Systems (GDS)" may be acceptable; however, this information should be supplemented to identify tests of the radioactivity control instrumentation. Also, additional discussion should be provided to cover beyond design basis conditions.

Response

The transient and accident scenarios carried out on the plant-specific Nine Mile Point Unit 1 (NMP1) simulator for performance testing of the SPDS encompass many of those previously used for validating the plant's new symptomatic Emergency Operating Procedures (EOPs). Categories of these events include the following:

- Loss of coolant accidents
  - Large break inside primary containment
  - Small break inside primary containment
  - Main steam line break outside primary containment
  - Unisolable breaks
  - Stuck open relief valves

- Loss of makeup flow to the reactor
  - Loss of feedwater
  - Failure of ECCS system(s) during loss of coolant accidents
  - Loss of all injection to the reactor vessel

- Loss of electrical power
  - Turbine-generator trip
  - Emergency diesel generator failure
  - Loss of off-site 345 kV power

- ATWS
  - Failure of the Reactor Protection System to initiate a reactor scram following receipt of a valid scram signal; manual scram required
  - Failure of control rods to insert when a scram is initiated by RPS (manual and automatic)

- Fuel element failure resulting in the off-site release of radioactivity

- Loss of primary containment integrity

- Loss of secondary containment integrity

Additional complications regarding equipment operability were imposed as appropriate to maximize the response (magnitude and rate of change) of various SPDS parameters, such as:

- Loss of drywell cooling, to maximize drywell temperature and pressure;

- Loss of torus cooling, to maximize heatup of the water in the torus;

- Failure of main turbine bypass valves and/or electromatic relief valves, to maximize reactor pressure.

The above types of events were conducted both singly and in combination such that many of the scenarios resulted in plant conditions significantly beyond the design basis events described in the FSAR. Multiple failures were initiated both sequentially and concurrently in order to achieve the conditions described. Most of the transients required that operators participating in the exercises had to enter and execute more than one EOP at a time in order to correctly respond to event conditions.

To the extent possible plant conditions were purposely manipulated to reach/exceed action levels and limits specified in the EOPs, and to facilitate observation of the widest possible range of SPDS display presentation changes (scales on bar graphs, display feature color coding,

indications of alarm status, indications of the direction of a parameter's rate of change, etc.). As a result, scenarios exercised parameters as follows:

- RPV Water Level - below the top of the active fuel, and above the main steam lines;

- RPV Pressure - above the lifting pressure setpoint of electromatic relief valves;

- Reactor Power - above the rated flow scram setpoint;

- Torus Water Temperature and RPV Pressure (in combination) - approaching the Heat Capacity Temperature Limit (the Limit at which Emergency RPV Depressurization is required by the EOPs);

- Drywell Pressure - approaching the Drywell Pressure Limit (the Limit at which venting of the primary containment to atmosphere is required by the EOPs);

- Drywell Temperature - above design temperature;

- Coolant Activity and Off-Site Radioactivity Release Rate - approaching that requiring declaration of an Alert condition in accordance with the site Emergency Plan.

The detailed outlines of individual scenarios used during the execution of the simulator exercises will be available for review in the project files.

2.4 <u>Human Factors Program</u>

The licensee should provide further clarification regarding how the proposed design will fulfill the NUREG-0737 Supplement 1, requirement for continuous display of plant safety status.

Response

A hierarchical display system is provided for the control room operators on a dedicated CRT. The primary level display (overview) provides key information on each of the five critical safety functions. Operators can quickly and easily access secondary level displays which provide supplemental information associated with each of the five critical safety functions. Visual cues are provided by the system to alert the operator of the status of all critical safety functions while viewing any display.

Plant procedures have been revised requiring the SPDS to be displayed at all times. Should any portion of the SPDS become inoperable, the computer department will be contacted and directed to repair the system as soon as possible.

2.5 Electrical and Electronic Isolation

The Safety Evaluation Report discusses the function, use and qualification testing of isolators provided by Rochester Instrument Systems. Although a response to this section was not required, Niagara Mohawk is submitting the following information for clarity and completeness.

Response

During the 1986 Refueling and Maintenance Outage additional inputs to the process computer were installed specifically for the SPDS. The isolation devices installed during this modification were manufactured by the Foxboro Instrument Company.

Foxboro Spec 200 Models N-2AO-VAI and N-2AI-T2V dual output and input modules were used. Both models are transformer isolated and qualified to IEEE 344-1975 and 323-1974. Input to output isolation can be maintained without internal damage when the following faults are applied:

1. Output shorted to +120 VAC.
2. Output shorted to power supply return or chassis ground.

[illegible] [illegible] [illegible]

3. Output disconnected from the load.
4. Line-to-line short between output terminals.

Additional information on these isolators is available in the project files. Similar isolators are used at Nine Mile Point Unit 2 and throughout the nuclear industry.