50-220

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

April 2, 1996

See Ltr.

Mr. B. Ralph Sylvia
Executive Vice President, Nuclear
Niagara Mohawk Power Corporation
Nine Mile Point Nuclear Station
P. O. Box 63
Lycoming, NY 13093

SUBJECT: NRC STAFF'S EVALUATION OF THE NINE MILE POINT NUCLEAR STATION UNIT NO. 1 INDIVIDUAL PLANT EXAMINATION (IPE) SUBMITTAL (TAC NO. M74436)

Dear Mr. Sylvia:

By letter dated July 27, 1993, as supplemented June 26, 1995, you responded to Generic Letter (GL) 88-20, "Individual Plant Examinations for Severe Accident Vulnerabilities," and Supplements 1, 2, and 3, thereto. With the assistance of our contractors, we have completed our review of the IPE submittal for internal events. The evaluation package consists of:

- The Staff Evaluation Report (SER) (Enclosure 1)
- The contractor's Technical Evaluation Reports (TERs) for the front-end, back-end, and human reliability analysis reviews (Enclosures 2, 3, and 4)

The Nine Mile Point Unit 1 submittal did not identify any severe accident vulnerabilities associated with either core damage or poor containment performance. We noted that as a result of the IPE or other industry initiatives, you implemented several procedural enhancements and hardware modifications which were reflected in your core damage frequency (CDF) estimate of 5.5E-6 per reactor year from internally initiated events, excluding internal flooding.

Based on our review of the Nine Mile Point Nuclear Station Unit No. 1 IPE submittal and associated documentation, we conclude that you have fully met the intent of GL 88-20.

GL 88-20 suggested that licensees could use their IPE submittals to address, among other safety issues, Unresolved Safety Issue (USI) A-45, "Shutdown Decay Heat Removal Requirements." In your response to GL 91-06, you had proposed to use the IPE submittal to respond to Generic Issue A-30, "Adequacy of Safety-

AA3

DF01

Related DC Power Supplies." These two generic issues are adequately resolved for Nine Mile Point Nuclear Station Unit No. 1 by your IPE submittal.

If you have any comments or questions, please contact me at (301) 415-3049.

Sincerely,

Original signed by:

Darl S. Hood, Senior Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-220

Enclosures: 1. Staff Evaluation
2. TER (Front-End)
3. TER (Back-End)
4. TER (Human Reliability Analysis)

cc w/encl 1: See next page

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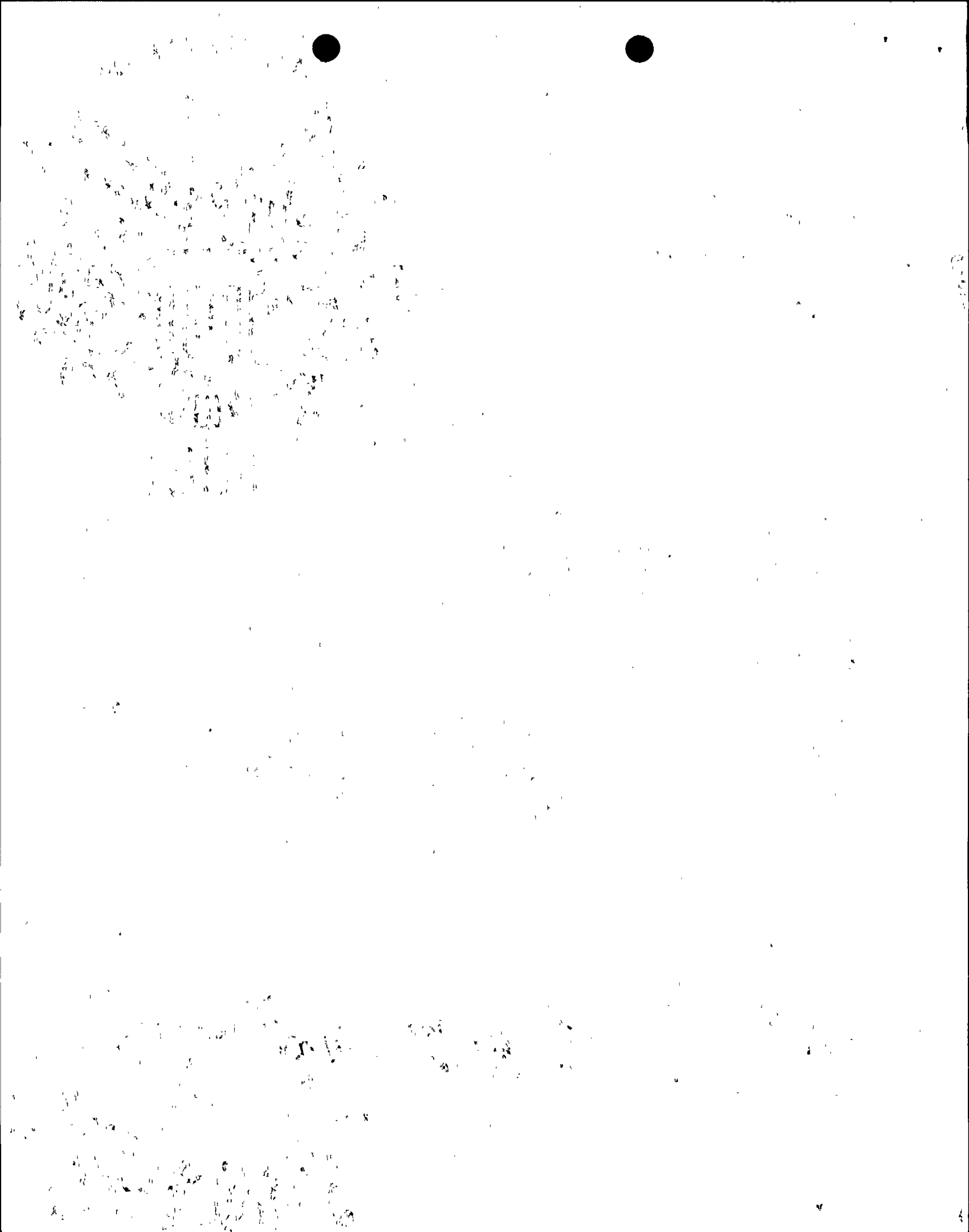

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DARL HOOD

Darl S. Hood, Senior Project Manager
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NINE MILE POINT NUCLEAR STATION UNIT NO. 1

INDIVIDUAL PLANT EXAMINATION

STAFF EVALUATION REPORT

Enclosure 1

1.0 INTRODUCTION

On July 27, 1993, Niagara Mohawk Power Corporation (NMPC or the licensee) submitted the results of the individual plant examination (IPE) for Nine Mile Point Nuclear Station Unit No. 1 (NMP1) in response to Generic Letter (GL) 88-20 and associated supplements. On April 21, 1995, the NRC staff sent questions to the licensee requesting additional information. The licensee responded in a letter dated June 26, 1995.

A "Step 1" review of the NMP1 IPE submittal was performed and involved the efforts of Science & Engineering Associates, Inc. (SEA), Scientech, Inc., and Concord Associates in the front-end, back-end, and human reliability analyses (HRA), respectively. The Step 1 review focused on whether the licensee's method was capable of identifying vulnerabilities. Therefore, the review considered (1) the completeness of the information and (2) the reasonableness of the results given the design, operation, and history of NMP1. A more detailed review, a "Step 2" review, was not performed. A summary of contractors' findings and the staff's evaluation is provided below. Details of the contractors' findings are in the technical evaluation reports (Enclosures 2, 3, and 4) attached to this staff evaluation report (SER).

In accordance with GL 88-20, the licensee proposed to resolve Unresolved Safety Issue (USI) A-45, "Shutdown Decay Heat Removal [DHR] Requirements." The licensee also proposed to resolve GI A-30, "Adequacy of Safety Related DC Power Supplies," as part of the NMP1 IPE. No other specific USIs or GIs were proposed for resolution.

2.0 EVALUATION

The NMP1 plant is a boiling-water reactor (BWR) 2 with a Mark I containment. In the IPE, the licensee has estimated a total core damage frequency (CDF) of 5.5E-6/reactor year excluding flooding. This CDF is lower than the CDFs for most BWRs. It appears that this low CDF estimate is driven by a low relative CDF contribution of transient events (8E-7/reactor year); the average transient relative CDF for BWRs is 8E-6. Station blackout contributes 64% transients 14%, loss-of-coolant accidents (LOCAs) 13%, and anticipated transients without scram (ATWS) 10%. Internal flooding was screened from the analysis on the basis of semi-quantitative flood scenario evaluations.

The important system/equipment contributors to the estimated CDF that appear in the top sequences are: failure to recover ac power, loss of ac and dc power, failure of the electromatic relief valves to reclose, and failure of the diesel fire pump to supply the reactor pressure vessel. It appears that the significant initiating events and dominant accident sequences were examined in the NMP1 Level 1 analysis.

Based on the licensee's IPE process used to search for DHR vulnerabilities and a review of NMP1 plant-specific features, the staff finds the licensee's DHR evaluation consistent with the intent of the resolution of USI A-45.

The licensee performed an HRA to document and quantify potential failures in human-system interactions and to quantify human-initiated recovery of failure events. The licensee identified the following operator actions as important in the estimate of the CDF: ac power recovery, emergency diesel load shedding

under LOCA conditions, reactor pressure vessel depressurization, prevention of the emergency (isolation) condenser (EC) isolation and EC recovery after isolation, core spray injection permissive calibration, feedwater control given loss of instrument air, dc load shedding given station blackout, and containment spray alignment for torus cooling mode.

The licensee evaluated and quantified the results of the severe accident progression through the use of a containment event tree and considered uncertainties in containment response through the use of sensitivity analyses.

Early releases (in less than 6 hours from accident initiation) occur 26% of the time, intermediate (between 6 and 24 hours) 48% of the time, and late (after 24 hours) 14% of the time; the containment remains intact 13% of the time. The staff noted that the licensee's definition of early containment failure (in less than 6 hours from accident initiation) was different from the typical definition of 2 hours from accident initiation. The licensee considered large releases as an indicator of containment performance. Large releases are defined in terms of accident sequences that will result in high releases (greater than 10% in CsI fission products) and early releases. Large release represents 13% of the total release frequency and is dominated by wetwell overpressure mainly occurring in ATWS sequences. The staff noted that although ATWS is not an important contributor to the overall CDF (10%), it is an important contributor to large releases. It appears that the important severe accident phenomena were considered in the NMP1 Level 2 analysis. The licensee's response to containment performance improvement (CPI) program recommendations is consistent with the intent of GL 88-20 and associated Supplement 3.

Some insights and unique plant safety features identified at NMP1 are:

1. The ECs do not initially require electrical power to provide core cooling thus extending the time for AC recovery during station blackout.

2. A hardened containment vent provides a backup to loss of containment cooling.

3. Eight-hour battery lifetime is relatively long compared to battery lifetimes at other BWRs and increases the likelihood of recovering offsite power.

4. The diesel-driven firewater pump provides makeup to the ECs; and

5. The capability to power the control rod drive pumps with the diesel generators ensures an additional source of makeup to the vessel even if offsite power is lost.

The licensee did not define what constitutes a plant vulnerability to severe accidents. It is stated in its submittal that no unusual or unique contributors to core damage nor unusually poor containment performance have been identified. However, the licensee implemented the flowing improvements:

1. Hardened vent.

2. Revision 4 of the BWR Owners Group emergency procedure guidelines; and

3. Initiation of the use of cross-tie containment spray raw water to core spray as an alternative source of injection to the reactor pressure vessel; initiation of the option align the containment spray raw water to the torus in order to flood the containment.

The licensee also identified several potential improvements for future use. The most important are:

1. Shedding the non-safety battery load so that it would be available to extend dc power supply after the safety batteries have failed during station blackout or using of a portable battery charger.

2. Improved calibration of low vessel pressure emergency core cooling system permissive sensors.

3. Capability to locally operate certain air-operated valves upon loss of instrument air.

4. Increased drywell head preload to improve containment integrity at elevated temperatures.

5. Modification of containment venting pressure in order to have high confidence that there is no large structural failure; and

6. Improved operator training in areas where the IPE took credit for human error recovery. Specifically, for recovering from loss of screenhouse intake, instrument air, and service water, and for controlling EC overfill events and operating the ECs after waterhammer events upon EC isolation.

3.0 CONCLUSION

On the basis of these findings, the NRC staff notes that: (1) the licensee's IPE is complete with regard to the information requested by GL 88-20 (and associated guidance in NUREG-1335), and (2) the IPE results are reasonable given the design, operation, and history of NMP1. As a result, the staff concludes that the licensee's IPE process is capable of identifying the most likely severe accidents and severe accident vulnerabilities and, therefore, the NMP1 IPE has met the intent of GL 88-20.

It should be noted that the staff's review primarily focused on the licensee's ability to examine NMP1 for severe accident vulnerabilities. Although certain aspects of the IPE were explored in more detail than others, the review is not intended to validate the accuracy of the licensee's detailed findings (or

quantification estimates) that stemmed from the examination. Therefore, this SER does not constitute NRC approval or endorsement of any IPE material for purposes other than those associated with meeting the intent of GL 88-20.

Principal Contributors: R. Clark
E. Lois

Date: April 2, 1996

NINE MILE POINT NUCLEAR STATION UNIT NO. 1

INDIVIDUAL PLANT EXAMINATION

TECHNICAL EVALUATION REPORT

(FRONT-END)

Enclosure 2