Docket File

UNITED STATES
NUCLEAR REGULATORY COMMISSION
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August 18, 1994

Docket No. 50-410

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

Dear Mr. Sylvia:

SUBJECT: INDIVIDUAL PLANT EXAMINATION FOR NINE MILE POINT NUCLEAR STATION, UNIT 2 (TAC NO. M74437)

On November 23, 1988, the NRC issued Generic Letter (GL) 88-20 which requires licensees to conduct an Individual Plant Examination (IPE) in order to identify potential severe accident vulnerabilities at their plant, and report the results to the Commission. Through the examination process, a licensee is expected to: (1) develop an overall appreciation of severe accident behavior; (2) understand the most likely severe accident sequences that could occur at its plant; (3) gain a more quantitative understanding of the overall probabilities of core damage and fission product releases; and, (4) if necessary, reduce the overall probability of core damage and radioactive material releases by modifying, where appropriate, hardware and procedures that would help prevent or mitigate severe accidents.

As stated in Appendix D of the IPE submittal guidance document NUREG-1335, all IPEs are to be reviewed by NRC teams to determine the extent to which each licensee's IPE process met the intent of GL 88-20. The IPE review itself is a two step process; the first step, or "step 1" review, focuses on completeness and the quality of the submittal. Only selected IPE submittals, determined on a case-by-case basis, will be investigated in more detail under a second step or "step 2" review. The decision to go to a "step 2" review is primarily based on the ability of the licensee's methodology to identify vulnerabilities, and the consistency of the licensee's IPE findings and conclusions with previous probabilistic safety assessment (PSA) experience. A unique design may also warrant a "step 2" review to better understand the implication of certain IPE findings and conclusions. The NRC staff's conclusions regarding the Nine Mile Point Nuclear Station, Unit 2 (NMP-2) IPE are based on a "step 1" review and are contained in our staff evaluation (Enclosure 1). The staff's review was performed with the assistance of Science and Engineering Associates, Inc., Scientech, Inc., and Concord Associates, Inc. (see Enclosures 2, 3, and 4, respectively).

By letter dated July 30, 1992, as supplemented by letters dated May 6, 1993, and July 14, 1994, which were in response to our requests for additional information dated March 9, 1993, and June 9, 1994, respectively, Niagara

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Mohawk Power Corporation (NMPC) submitted the NMP-2 IPE in response to GL 88-20 and associated supplements. The IPE submitted is based on an internal events level 2 assessment consistent with the guidance provided in GL 88-20, Appendix 1. The IPE process also addressed internal flooding. NMPC plans to provide a separate submittal for external events (IPEEE), which will be reviewed separately within the framework prescribed in GL 88-20, Supplement 4.

NMPC's IPE analysis of NMP-2, a BWR 5 with Mark II containment, obtained a mean core damage frequency (CDF) of 3.1E-05/yr. This is a factor of six higher than that estimated for either of the BWRs (Peach Bottom Unit 2 or Grand Gulf) in the NUREG-1150 study. The NUREG-1150 BWR Probabilistic Risk Assessments (PRAs) are dominated by station blackout (SBO) sequences, while the NMP-2 PSA is dominated by transients. According to the NMP-2 IPE, transients contribute about 77 percent to the total CDF, SBO 17 percent, anticipated transients without scram (ATWS) 3.5 percent, and loss-of-coolant accidents (LOCAs) 2.5 percent. Among the transients, loss of divisional AC power (emergency or offsite) contributes 31 percent and loss of either division of 115 kV contributes 15 percent. The initiation frequencies for loss of AC power initiators are lower in comparison to those for other plant trip initiating events, but the plant challenges and consequences are more severe.

The NMP-2 IPE showed that loss of one divisional AC power train (e.g., due to switchgear failure) is perhaps a more important initiating event than loss of offsite power (LOSP), since, although less likely (initiation frequency, IF=0.0043/yr) than LOSP (IF=0.04/yr), most of the other support systems as well as front-line systems (i.e., all class 1E safety related equipment) depend on emergency or offsite AC power to operate. Loss of one division of AC power disables all safety systems which depend on it and causes all the division's service water pump breakers to open which in turn isolates the Reactor Building Closed Loop Cooling (RBCLC) and Turbine Building Closed Loop Cooling (TBCLC) systems. Loss of cooling to the condenser, feedwater, and turbine generator equipment requires an immediate shutdown by the operators. Isolation of the RBCLC and TBCLC systems is assumed to result in a low flow trip of the opposite division's service water pumps, which then must be restarted. The dominant sequences for this initiator involve: a) failure on demand of the opposite division's battery preventing restart of the division's service water pumps and start of the division's safety systems, such as reactor pressure vessel (RPV) injection, or b) independent failure of the Service Water System which causes loss of condenser and feedwater as well as room cooling, resulting in the loss of High Pressure Core Spray (HPCS) and Reactor Core Isolation Cooling (RCIC). No credit is taken for recovery of failed equipment associated with loss of divisional AC power and service water over the 20 to 30 hour time frame associated with the containment failure. NMPC is developing procedures for preventing RCIC trip under loss of service water conditions.

NMP-2 has 3 onsite emergency diesel generators, one for each of the two divisional AC Power Trains and the third dedicated to the HPCS System. Loss of offsite AC power disables normal operating non-safety systems such as the

condenser, feedwater, RBCLC, TBCLC, and instrument air, as well as normally operating safety systems such as the Service Water System which must restart on demand after the emergency diesels start and load. The dominant LOSP sequence involves the unavailability of all emergency diesel generators (with loss of HPCS due to loss of service water), inability to recover offsite power or emergency diesels within 30 minutes, and unavailability of RCIC, resulting in station blackout and core damage. No credit is given to aligning the diesel fire pump within the first 2 hours of a SBO because of the time required to perform these actions and the potential for insufficient flow to the RPV.

The 10 core damage sequences that make the largest contribution to the CDF together amount to about 40 percent of the total. A ranking of systems for the prevention of core damage in relative order of importance yielded Emergency AC Power, Residual Heat Removal (RHR), Containment Vent, HPCS, Service Water System, RCIC, and Emergency DC Power, respectively. Similarly, an ordering of operator actions according to their importance to core damage prevention yielded AC power recovery, containment venting, emergency depressurization, operation of Service Water System, and Emergency Core Coolant System pump room cooling, respectively.

Each containment event tree sequence ended in one of 13 end-state/source term categories which are defined in terms of severity (e.g., high or H for CsI releases greater than 10 percent) and time (e.g., early or E for releases up to 6 hours after accident initiation) of release. Only CsI was considered as the source term. The total frequency of release was estimated to be 2E-05/yr, with H/E releases contributing 3 percent. The dominant containment failure modes were found to be failure of the drywell head (58 percent) and failure of the wetwell (vapor space, 15 percent, and below the water line, 9 percent).

Noteworthy among the unique features of NMP-2 listed by NMPC are: a) a redundant reactivity control system which automatically actuates standby liquid control (SLC), reactor recirculation pump trip, alternate rod insertion, and feedwater runback, and b) a hardened Containment Vent System with appropriate emergency operating procedures. As a result of the IPE, NMPC planned to make the following improvements to plant configuration and operating procedures: installation of valves in the Standby Gas Treatment System to increase the reliability of the Containment Venting System; development of procedures to enhance Auxiliary Bay pump room cooling during loss of service water accidents to reduce the likelihood of loss of HPCS, RCIC, and low pressure injection pumps; enhancement of station blackout procedures (for which credit was taken in the IPE evaluation of SBO scenarios); enhancement of procedures for dealing with internal flood scenarios; and improved test and maintenance procedures to reduce the likelihood of an interfacing system LOCA. However, NMPC has not implemented a proposed containment vent modification and five procedural enhancements for which credit was taken in the IPE. If these changes are not implemented, NMPC should revise the IPE to reflect the as-built, as-operated plant. As a minimum, the revision should include a revised list of sequences that meet the GL 88-20 or NUREG-1335 screening criteria and their associated contribution to

the NMP-2 CDF. In addition, any initiators, human actions, or common cause failures that have been modified, added or eliminated because of the revision, and an evaluation of the need for any additional plant improvements to address potential vulnerabilities, should be included. These revisions need not be submitted to the NRC but should be retained in the plant records for future inspections if requested by the NRC.

Other significant insights are presented in the Summary of the Nine Mile Point Nuclear Station, Unit 2, IPE Submittal on Internal Events (Enclosure 5).

No specific unresolved safety issues (USIs) or generic safety issues (GSIs) were proposed for resolution as part of the NMP-2 IPE.

This concludes the NRC staff's review efforts associated with TAC No. M74437. We conclude that NMPC has met the intent of GL 88-20.

Sincerely,

Pao Tsin Kuo, Acting Director
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. NRC staff evaluation of NMP-2 IPE
2. TER for front-end analysis
3. TER for back-end analysis
4. TER for human reliability analysis
5. Summary of the NMP-2 IPE submittal on internal events

cc w/enclosures 1-5:
See next page

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ENCLOSURE 1

NRC STAFF EVALUATION OF THE NINE MILE POINT NUCLEAR STATION, UNIT 2 INDIVIDUAL PLANT EXAMINATION

(IPE)

(INTERNAL EVENTS ONLY)

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EXECUTIVE SUMMARY

The NRC staff completed its review of the internal events portion of the Nine Mile Point Nuclear Station, Unit 2 (NMP-2); individual plant examination (IPE) submittal, and associated documentation which includes licensee responses to staff generated questions and comments. Niagara Mohawk Power Corporation's (NMPC's) IPE is based on a full scope level 2 Probabilistic Risk Assessment (PRA) performed in fulfillment of Generic Letter (GL) 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities - 10 CFR 50.54(f)," and is documented in the submittal. No specific Unresolved Safety Issues (USIs) or Generic Safety Issues (GSIs) were proposed for resolution as part of the IPE.

NMP-2 is a BWR 5 plant with a Mark II containment. The IPE estimates the mean core damage frequency (CDF) as 3.1E-5/yr. Contributions from some important initiating events are as follows: Loss of Offsite Power (LOOP) contributes 26 percent (blackout 17 percent and non-blackout 9 percent), loss of either division of emergency AC power contributes 31 percent and partial loss of offsite power (loss of either division of 115 kV) 15 percent. The licensee also identified contributions from functional groupings. A large fraction of the CDF is associated with sequences which are contained in functional groupings such as loss of injection 50 percent (non-SBO), loss of heat removal 29 percent, and SBO 18 percent. In addition, the submittal also provides a discussion of the top 10 highest frequency sequences which account for about 40 percent of the total CDF with the first three sequences (LOOP, loss of division 1, 2) contributing 23 percent. No other individual sequence contributes greater than 3 percent to the overall CDF.

The licensee did not provide a definition of vulnerability in the submittal but in response to questions indicated that assessment of plant vulnerabilities was made using a screening process. The first criterion of the screening process was CDF greater than 1E-4 per year or early release greater than 1E-6 per year. The licensee did not identify any severe accident vulnerabilities associated with either core damage or containment failure using this process. However, the licensee indicated that consideration of plant improvement initiatives was not limited to this process and that a detailed review of the model and the results for areas where improvement initiatives could be warranted was performed. The licensee probed the results by performing importance analyses for top events, split fractions, and operator actions. As a result of this review, the licensee identified one hardware modification and five procedural enhancements (intended to reduce the probability of human errors) for which credit has been taken in the IPE. These improvements, which focused on both reducing CDF and offsite release of radioactivity were scheduled for implementation by the end of the 1993 refueling outage but they have not been implemented.

Based on the review of the NMP-2 IPE submittals and associated documentation, the staff concludes that the licensee met the intent of GL 88-20. This conclusion is based on the following findings: (1) the IPE is complete with respect to the information requested in GL 88-20 and associated Supplement 3; (2) the analytic approach is technically sound and capable of identifying plant-specific vulnerabilities, including those associated with internal flooding; (3) the licensee employed a viable means to verify that the IPE models reflect the current plant design and operation at time of submittal to the NRC; (4) the IPE had been peer reviewed; (5) the licensee participated in the IPE process; (6) the IPE specifically evaluated the decay heat removal function for vulnerabilities; (7) the licensee responded appropriately to Containment Performance Improvement (CPI) program recommendations. In addition, NMPC has indicated that "as additional information and technology becomes available the IPE will be extended, updated and used by NMPC, and that based on the IPE and its updates, improvement initiatives will continue and the IPE as a living program, will continue to benefit the plant."

I. BACKGROUND

On November 23, 1988, the NRC issued GL 88-20 which requires licensees to conduct an IPE in order to identify potential severe accident vulnerabilities at their plant, and report the results to the Commission. Through the examination process, a licensee is expected to: (1) develop an overall appreciation of severe accident behavior, (2) understand the most likely severe accident sequences that could occur, (3) gain a more quantitative understanding of the overall probabilities of core damage and fission product releases, and (4) if necessary, reduce the overall probability of core damage and radioactive material releases by modifying, where appropriate, hardware and procedures that would help prevent or mitigate severe accidents.

As stated in Appendix D of the IPE submittal guidance document NUREG-1335, all IPEs are to be reviewed by the staff to determine the extent to which each licensee's IPE process met the intent of GL 88-20. The IPE review itself is a two step process; the first step, or "Step 1" review, focuses on completeness and the quality of the submittal. Only selected IPE submittals, determined on a case-by-case basis, will be investigated in more detail under a second step or "Step 2" review. The decision to go to a "Step 2" review is primarily based on the ability of the licensee's methodology to identify vulnerabilities, and the consistency of the licensee's IPE findings and conclusions with previous PRA experience. Unique designs may also warrant a "Step 2" to better understand the implication of certain IPE findings and conclusions. As part of this process, the NMP-2 IPE only required a "Step 1" review.

On June 30, 1992, NMPC (the licensee for NMP-2) submitted the NMP-2 IPE in response to GL 88-20 and associated supplements. NMP-2 is a BWR 5 plant with a Mark II containment. The IPE submittal described the application of a Level 2 PRA to identify vulnerabilities, consistent with GL 88-20. The IPE submittal contains the results of an evaluation of internal events, including internal flooding. The licensee plans to provide a separate submittal on findings stemming from the IPE external events analysis (IPEEE). The staff will review the IPEEE separately, within the framework prescribed in GL 88-20, Supplement 4. On March 9, 1993, and June 9, 1994, the staff sent requests for additional information to the licensee. The licensee responded to the staff's requests in letters dated May 6, 1993, and July 14, 1994, respectively. Information reviewed by the staff during the IPE process evaluation included the IPE submittal and the licensee's response to staff questions. In addition, the staff contracted Science & Engineering Associates (SEA) Incorporation to review the Level 1 analysis; Scientech Incorporation and Energy Research Incorporation to review the Level 2 analysis; and Concord Associates to review the human reliability analysis. SEA's review is documented in SEA 92-553-07-A:1, "Nine Mile Point 2 IPE: Front-End Audit." Scientech's review is documented in SCIE-NRC-211-92, "Step 1 Technical Evaluation Report of the Nine Mile Point 2 Individual Plant Examination Backend Submittal." Concord's review is documented in CA/TR-92-019-07, "Technical Evaluation Report: Nine Mile Point Unit 2 Individual Plant Examination Assessment of Human Reliability Analysis Step 1 Review."

II. NRC STAFF'S REVIEW

1. Licensee's IPE Process

The NMP-2 IPE submittal describes the method by which the licensee confirmed that the IPE represents the as-built as-operated plant. A key element is the high degree of involvement in the IPE by the NMPC staff who brought a knowledge of plant operations to the analysis. In addition to using significant sources of information regarding the plant configuration and procedures to document plant status for the analysis, a number of plant walkdowns of the containment, the reactor building and individual systems were performed with personnel from the plant staff, the IPE group and independent consultants. The system descriptions identify the references (drawings and procedures) used for the system information and analysis. The event trees reference the emergency operating procedures (EOPs).

The IPE submittal contains a summary description of the licensee's IPE process, the licensee's participation in the process, and the subsequent in-house peer review of the final product. NMPC organized a team under the responsibility of the Nuclear Technology Department to perform the IPE and to ensure that NMPC personnel were involved in all aspects of the IPE. Five full time engineers were assigned to the PRA team. They were supported by a 20-member team from various departments within the organization. In addition consultants from Fauske & Associates, Gabor, Kenton & Associates, ABB Impell, Erin Engineering, Halliburton NUS, and General Physics assisted in thermal hydraulic analysis, containment performance, and human reliability analysis.

A separate review team under the direction of the Quality Assurance Department and Independent Safety Evaluation Group was formed to independently ensure technical accuracy. NMPC indicated that "as additional information and technology becomes available the IPE will be extended, updated and used by NMPC, and that based on the IPE and its updates, improvement initiatives will continue and the IPE as a living program, will continue to benefit the plant."

The licensee did not provide a definition of vulnerability in the submittal but in response to questions indicated that an assessment of plant vulnerabilities was made using a screening process. The first criterion of the screening process was CDF greater than 1E-4 per year or early release greater than 1E-6 per year. However, the licensee indicated that consideration of plant improvement initiatives was not limited to this process, and that a detailed review of the model and the results for areas where improvement initiatives could be warranted was performed.

From the results of the analysis, the licensee identified one hardware modification and five procedural enhancements proposed for completion by the end of the refueling outage scheduled for December 1993. However, these proposed changes have not been implemented.

2. Front-end Analysis and DHR Evaluation

The staff examined the front-end analysis as described in the IPE submittal for completeness and consistency with accepted Probabilistic Safety Assessment (PSA) practices. In response to GL 88-20, NMPC has performed an Internal Events Level 2 PRA for their IPE. The front-end IPE analysis used the "Large Event Tree, Small Fault Tree" modeling technique (an approach described in NUREG/CR-2300). Fault trees were used to quantify system failure values which were used as input to the event tree nodes. Logic models for the fault trees were developed using the CAFTA computer code. The RISKMAN computer code was used for quantification of the CDF.

The complete level 2 analysis was accomplished by directly linking the front-end event tree and the containment event tree (CET) through interface rules defining plant damage states. The information carried from the level 1 to the level 2 analysis accounts for preexisting conditions that would impact the back-end analysis and is consistent with other PSAs.

The licensee's process identified 29 initiating events categories for NMP-2 which are captured in 4 broad groups: loss of coolant inventory, transients, support systems transients and internal floods. Three of these events are internal flooding events. FMEAs and fault trees of plant systems were used to identify plant specific initiators. ATWS events were not defined as a separate initiating event, but instead were addressed through the development of a special event tree for all initiating events that are followed by a failure of reactor trip. The staff has compared and found the list of initiators consistent with lists from other PSAs and NUREG/CR-2300.

Systemic event trees were developed for each unique initiating event group. The IPE submittal contained all frontline and support system event trees, and special trees, developed to address the plant response to transients, station blackout, LOCAs, and ATWS events. System success criteria were presented for each initiating event category. The licensee has stated that system success criteria are based on the overall success criteria for onset of core damage, containment integrity and reactor pressure vessel integrity; the basis of which are BWR generic (owners group and G.E.) and plant-specific analyses using the MAAP code. In general the staff finds the NMP-2 event trees and special trees to be consistent with regard to initiating events, associated success criteria, and dependencies between top events.

The IPE submittal addressed dependencies by providing dependency matrices which identified support to support and support to front-line dependencies on a train basis, and by incorporating the dependency impacts in the logic which assigns split-fractions to the nodes in the event trees. A separate screening evaluation was done for the HVAC systems to determine their impact on important safety equipment. Loss of HVAC to the diesel generator control room, the HPCS pump room and the north and south auxiliary bay MCC rooms were found to be important. The licensee incorporated the first two into the system model and the MCC rooms into the support system event tree. The licensee also indicated that heatup calculations were not performed for areas which were large and open in comparison to the heat load. This included the essential switchgear rooms. The licensee indicated that there would be "adequate" time for operator diagnosis and recovery action for these areas.

The NMP-2 PRA Model was quantified with generic data from PLG Inc.'s Database for Probabilistic Risk Assessment of Light Water Nuclear Power Plants (PLG-500) and the Nuclear Computerized Library for Assessing Reactor Reliability (NUCLARR). Plant and unit specific data was obtained from the NMP-2 Inservice Testing Database, the NMP-2 Nuclear Plant Reliability Database, and the NMP-2 INPO Quarterly Performance Indicator Data (maintenance unavailability). Plant specific data was combined with generic data through use of Bayesian updating techniques. In addition, the licensee indicated that it grouped data for plant specific components such as the ECCS pumps (except RCIC). The initiating events involving system failures were developed using plant specific system analyses.

The IPE has considered the impact of common cause failures (CCF) due to system dependencies by incorporating them explicitly in the event tree logic. Additional component CCF were addressed through the use of plant-specific component CCF failure factors. The methodology used for quantification of CCF failure factors for the NMP-2 IPE submittal was the Multiple Greek Letter Method. The staff notes that the licensee's analytic treatment of CCF is consistent with NUREG CR-2300 and NUREG CR-4780.

The licensee's IPE flood analysis employed a screening analysis to determine potential flood sources, locations, propagation paths, impacts on plant operation, and the ability of the operations staff to safely shut down the plant. All flood initiators except three were screened out; two in the diesel room and one in the service water pumphouse. The contribution from internal floods was estimated to be 5 percent of the CDF. The dominant sequence is a service water flood (3 percent), in the emergency diesel room that is not isolated for 2 hours. The licensee indicated that maintenance activities were considered in the quantification of all postulated internal flood sequences. In response to staff questions, the licensee also indicated that it relied on existing design basis analyses for assessment of events with potentially damaging effects from water intrusion due to impingement, spray, or splashing. For example, the FSAR medium energy line break analysis indicates that critical equipment required to ensure the safe shutdown of the plant are adequately protected from any postulated breach in a water line (causing spraying or splashing onto the equipment) in these areas. In response to questions, the licensee also indicated that even though operators have longer than 1.5 hours to isolate a flood source or open doors to prevent flooding, additional direction to mitigate flooding has been added to existing procedures to provide additional margin.

The submittal identified the 100 highest frequency sequences in accordance with the reporting guidelines in NUREG-1335. The IPE estimated the mean core damage frequency as 3.1E-5/yr. The results of the analysis were expressed in various classes of accident sequences groupings, e.g., initiating events, classes of specific interest (e.g., loss of injection), plant damage states (PDS) and individual sequences. Contributions from important initiating events include: Loss of Offsite Power (LOOP) contributes 26 percent (blackout

17 percent and non-blackout 9 percent), loss of either division of emergency AC power contributes 31 percent and partial loss of offsite power (loss of either division of 115 kV) 15 percent. For sequences of specific interest, a large fraction of the CDF is associated with sequences which contain loss of injection 68 percent (SBO 18 percent, non-SBO 50 percent), and loss of heat removal 29 percent. In addition, the submittal also provides a discussion of the top 10 highest frequency sequences, which account for about 40 percent of the total CDF with the first three sequences (LOOP, Loss of division 1, 2) contributing 23 percent. All additional sequences contribute less than 3 percent to the overall CDF.

In accordance with the resolution of USI A-45, the licensee has performed an evaluation of the NMP-2 DHR system as an intrinsic part of the NMP-2 IPE examination to identify DHR vulnerabilities. The results of the IPE indicated the importance of the systems that provide the decay heat removal function. This is measured by the percentage of CDF attributable to sequences that involve failure of the DHR top events and split fractions. The following systems performing decay heat removal functions were considered in the DHR evaluation:

- Main Condenser
- RHR System
- Containment Venting
- Continued Injection Following Heat Removal

The contribution of long-term heat removal systems to CDF is 29 percent. Systems important to the loss of decay heat removal include the RHR System (top events LA and LB; the contribution to CDF of the sequences containing these top events are 17 percent and 14 percent, respectively) and Containment Venting (top event CV; 17 percent). Support systems important to decay heat removal include service water (top events SA, 14 percent & SB, 12 percent) and emergency AC Division II (top event A2, 26 percent).

Based on the staff's review of the IPE front-end analysis and the finding that the employed analytical techniques are consistent with other NRC reviewed and accepted PSAs and capable of identifying potential core damage vulnerabilities, the staff finds that the NMP-2 IPE front-end analysis meets the intent of GL 88-20.

3. Back-End Analysis and Containment Performance Improvements (CPI)

The staff examined the licensee's back-end analysis for completeness and consistency with the guidance specified in GL 88-20, Appendix 1.

The NMPC consultant, PLG Incorporated, used the RISKMAN computer code to quantify the event trees. Version 8 of the MAAP-3.0B code was used. Due to two errors in Version 8, the licensee has rerun affected analyses with version 8.01 (which has corrected the errors). The results indicate that the information in the IPE submittal is still applicable. For example, the revised analysis indicated that the pool decontamination factor was larger than the value used in the IPE submittal. The analyses conformed to EPRI's recommendations related to selected model parameter values and suggested sensitivity studies to be considered. The licensee had ABB Impell perform a plant specific containment structural analysis to develop containment failure pressure, temperature, and location insights. The mean ultimate containment failure pressure was determined to be 141 psig.

The translation of the Level 1 accident sequences into Level 2 accident release characteristics was performed by mapping each of the accident sequences into one of 13 end-state/source term categories. These categories were defined by the severity and time of release. The severity of the release was based on the amount of CsI released, as follows:

- High (H) - Greater than 10 percent CsI released,
- Moderate (M) - Between 1 percent and 10 percent CsI released,
- Low (L) - Between 0.1 percent and 1 percent CsI released,
- Low-Low (LL) - Less than 0.1 percent CsI released, and
- No release/Negligible - Much less than 0.1 percent CsI released.

Only CsI was considered as the source term; that is, elemental iodine and other elements normally modeled in source term assessments were not considered in defining the amount of the release to the environment. The severity was assessed, independent of accident class, with the RISKMAN and MAAP codes. This included consideration of the potential containment failure location, the availability of sprays, and reactor building effectiveness. The timing of the release was based on the estimated containment failure time from the initiation of the accident, as follows:

- Early (E) - 0 to 6 hours,
- Intermediate (I) - 6 to 24 hours, and
- Late (L) - More than 24 hours.

Sensitivity studies were performed by varying MAAP values within allowable ranges. Issues which were not analyzed in sensitivity studies were considered to be categorized as high releases with early containment failure. By design, corium will not be retained inside the drywell, but will passed to the suppression pool floor via in-pedestal downcomers. This minimizes the core-concrete interactions and the attendant release of noncondensible gases.

The licensee considered the effects of containment temperature and pressure on the elastomer seals in the drywell and wetwell. These seals are used for the drywell head flange and equipment and manway hatches. For all of the potential accident sequences considered, the temperature and pressure profiles are expected to result in no or little leakage. This result is based on their consultant's analysis (ABB Impell) and is consistent with the results of analysis discussed in NUREG/CR-5565, NUREG/CR-4944, NUREG/CR-5096, and NUREG/CR-4064.

The modeling of containment isolation failure is based on a fault tree model. The fault tree incorporates modeling of containment hatches and large lines that penetrate containment and are open to the containment atmosphere (e.g., vent and purge lines). The fault tree considers automatic isolation signal failure, preexisting open pathways, manual isolation, and component failures. Failure of containment isolation is modeled as a failure in the drywell.

Containment isolation failure is characterized as a high release with early containment failure, based on a representative worst-case MAAP calculation.

The licensee's assessment involves the direct linking of each sequence from the Level 1 to the Containment Event Tree (CET) evaluation. This couples the front-end to the back-end portion of severe accident sequences through the directly linked event trees. In this way the dependencies are accounted throughout the front-end and back-end trees. The licensee has listed the top 100 Level 2 sequences, consistent with the NUREG-1335 screening guidelines.

The decontamination factor (DF) of the reactor building on the release has been considered as a sensitivity. Either the DF calculated by the MAAP code was used (reactor building assumed effective in removing radioactive material) or a DF = 1 was used (no removal of material in the reactor building). The suppression pool DF has been conservatively assumed to be 10 for a subcooled pool.

The Containment Performance Improvement program, made a number of recommendations that were expected to enhance containment performance. As a part of the IPE process licensees were to consider these recommendations in the IPE. These recommendations were identified in GL 88-20, Supplement 3, for Mark II plants. NMP-2 being a Mark II plant, was also expected to consider the Mark I improvements (included in Supplement 1 to GL 88-20) for applicability to Mark II containments. Each of these proposed improvements is discussed separately below.

1. <u>A hardened vent</u>: A hardened wetwell vent was to be installed during the 1993 refueling outage, however, this modification has not been installed. Venting of containment was to be initiated at 45 psig, prior to reaching the Primary Containment Pressure Limit (PCPL), as specified in the EOPs. The over pressure failure of containment in the TW (Class II) sequences has been estimated by the licensee to occur 31 hours after reactor scram and loss of containment heat removal capability. The conditional probability of containment failure given core damage due to operators opening containment vent is 4 percent. The licensee has properly modeled the vent in the PRA and has performed a sensitivity study which indicates that use of a higher venting pressure and cyclic operation of the vent valve, as compared to continuously open, can reduce the offsite releases by up to an order of magnitude. The licensee proposes to study this possibility further as part of its accident management program. Although the licensee did not explicitly address alternate means for containment heat removal, the licensee did perform sensitivity analyses on operation of containment venting, suppression pool mixing, drywell spray usage, containment flooding, and accident management actions. Thus, the licensee has considered alternate means for containment heat removal.

2. <u>An alternative water supply for vessel injection or drywell sprays</u>: Provisions for using the fire protection system pumps aligned to supply the RHR system have been provided. The EOPs instruct the operators to

use this capability for injection when RPV water level cannot be maintained above 159.3 inches. The fire protection system consists of one AC-powered pump and one diesel-driven pump, each having a capacity of 250 gpm. The diesel for the diesel driven pump has its own DC power supply and it can be started locally. Connection of the fire protection system to the Residual Heat Removal (RHR) system is by means of a hose connection from a fire line to the suction of the RHR pump. During normal operation, the connection at the RHR pump is closed by means of a blind flange. However the licensee has indicated in the submittal that, "There is uncertainty about the capability of the diesel fire water to provide successful injection to the reactor pressure vessel through the RHR system by way of 100 feet of 2.5 inch hose. Test data and additional information or tests are being pursued to establish a system injection flow profile."

3. <u>An enhanced reactor pressure vessel (RPV) depressurization system reliability</u>: NMP-2 has 18 safety relief valves (SRVs). The Automatic Depressurization System (ADS) which is a subset of the SRVs, consists of 7 SRVs, and is, by procedure, manually inhibited to prevent operation. Each SRV has its own nitrogen accumulator tank and redundant actuation solenoids, one from each DC Bus. If, as the operator is instructed by the EOPs, an SRV is opened once and left open, the valve will remain open for 15 hours without additional nitrogen makeup. Success is defined as the ability to maintain the RPV depressurized for 24 hours. Any two valves will depressurize the RPV. Therefore, using 7 SRVs manually (as modelled by the licensee), the RPV will remain depressurized for 15 hours. By using a different 7 SRVs or nitrogen make up (which is also modeled by the licensee), the RPV will remain depressurized for hours 16 through 30, with 4 additional valves unused. The IPE indicates that the only time the reactor cannot be depressurized is when the SRVs themselves fail. As a result, the licensee has concluded that additional depressurization reliability is not needed.

4. <u>Incorporation of the BWROG Emergency Procedure Guidelines (EPGs), Revision 4, into the plant procedures</u>: The licensee has incorporated Revision 4 of the BWROG EPGs.

Based on this review, the staff concludes that the licensee's response to the CPI Program recommendations, which included searching for vulnerabilities associated with containment performance during severe accidents, is reasonable and consistent with the intent of GL 88-20 and associated Supplement 3.

The licensee employed a process to understand and quantify severe accident progression. The process led to a determination of conditional containment failure probabilities and containment failure modes consistent with the intent of GL 88-20, Appendix 1. Sensitivity studies were performed. Failures of containment due to phenomenological considerations where uncertainties (such as direct containment heating, steam explosions, and hydrogen deflagrations) were considered to be high releases with early containment failure. A comparison with other Mark II containments is shown in the following table.

Conditional Containment Failure Probability

▪ NMP-2	69.0 percent (IPE)
▪ LaSalle	42.5 percent (RMIEP)
▪ Limerick	25.3 percent (IPE)

CONDITIONAL FAILURE PROBABILITY FOR TIMING

No failures	26 percent
Early/high	2 percent
Early/other	5 percent
Intermediate	36 percent
Late	30 percent

The dominant contributors to containment failure were found to be consistent with insights from other analysis of similar designs. The licensee characterized containment performance for each of the CET end-states. The overall assessment of the back-end analysis is that the licensee has made reasonable use of probabilisitic techniques in performing the back-end analysis, and that the techniques employed are capable of identifying plant vulnerabilities. Based on these findings, the staff concludes that the licensee's back-end IPE process is consistent with the intent of the GL 88-20.

4.0 Human Factor Considerations

The NMP-2 IPE submittal documents the human reliability assessment (HRA) methodology used for the front and back end analysis. The staff examined IPE HRA for completeness and consistency with acceptable PSA practices.

In performing the HRA, the licensee divided the human errors into two categories: pre-initiator events (errors that disable a system prior to a demand for their operation; such as may be made during test and maintenance) and post-initiator events (actions performed in responding to an accident). Generally the investigation of the pre-initiators relied strongly on industry data rather than on the HRA results. The licensee focused on human induced common cause failures that could potentially disable several trains of a system and thus fail a critical safety function.. This focus followed from the licensee's position that unavailability due to pre-initiator human failures in single train systems is small, and that component failure data quantitatively envelopes maintenance errors.

The licensee conducted a review of industry data (INPO LER data for BWRs) for events involving misalignment and miscalibration that were applicable to NMP-2. All events not of a common cause nature were screened out. No attempt to quantitatively estimate the impact of the elimination of the screened out events as a result of the licensee's position was made, nor was any discussion provided identifying how NMPC assured themselves that the data used captured unavailability due to pre-initiators. As a result of the screening process used in the review, only two pre-initiator type events were incorporated in the IPE; failure to restore the SLC system to the normal post-test configuration, and miscalibration of ECCS pressure instrumentation.

The model used for evaluation of Human Error Probabilities (HEPs) splits the response into two components, a detection, diagnosis, and decision phase (DDD), and an execution phase. Various methods were used in the IPE to quantify the HEPs. In the IPE submittal and the response to staff questions, the licensee indicated that the ASEP method (NUREG/CR-4772) was used for many of the HEPs in the DDD phase and for most of the HEPs in the execution phase. The EPRI approach (An Approach to the Analysis of Operator Actions in Probabilistic Risk Assessment, EPRI-TR-100259) was used for the DDD contribution for those Human Interactions (HIs) associated with missing a certain step in a procedure, and THERP (NUREG/CR-1278) was used for particularly important well practiced actions where it was known that ASEP would provide a conservative value. The HRA process identified significant operator functions from event sequences which were broken down to basic HI for quantitative analysis. The human actions were modeled as top events in the event trees and those actions that might impact only one system were modeled with the system as basic events in the fault trees. In response to staff questions, the licensee identified the type of Performance Shaping Factors (PSF) which were used in the analysis to modify the basic HEPs. Some PSFs, used in the EPRI approach were nature/clarity of cues, training on response, and quality of procedures. Among the PSFs, identified as being addressed explicitly in applying the ASEP and THERP methodologies, were stress, time, training, and crew structure. In addition, the licensee considered potential dependencies in the quantification of the HEPs which are reflected in the probabilities used. The groundrules for identification of cognitively related HIs were identified in the submittal.

In the IPE submittal and the response to staff questions, the licensee indicated that the information was obtained through review of plant procedures, interviews with operations and training staff, and talk and walk-throughs of operator responses at the simulator and in the plant. It was indicated that the <u>results</u> of these interactions, but not the structure used to capture them are in a human reliability analysis file.

The licensee did not provide the sequences that, but for low human error rates in recovery actions, would have been above the applicable screening criteria. However, in response to a staff question, the licensee provided information identifying the "importance" and "risk achievement worth" (the factor increase in CDF when in this case, the human error is set to guaranteed failure (1.0)) of all human actions in the IPE. A qualitative assessment was provided for human actions associated with the level II analysis. Of note are the risk achievement worth of the following human actions: operator fails to align RHR (CDF increases 279 times), operators fail to initiate RPV depressurization (CDF increases 92 times), operator fails to locally open RHR heat exchanger MOV (CDF increases ≈ 6.7 times), containment venting (CDF increases 4.3 times). Additional information regarding proposed procedural enhancements for other operator actions is contained in Section 5. The licensee also indicated that NMPC personnel were actively involved in all phases of the HRA and are capable of applying HRA-IPE techniques to emerging plant issues. However, the licensee stated that it would likely use outside consultants should updates be performed, due to workload and a desire to maintain the IPE as "state-of-the-art." The licensee has identified five procedural enhancements intended to reduce the probability of human errors for which credit has been taken in the IPE. (See Section 5).

Based on the staff's review of the licensee's IPE submittal and responses to staff questions, the staff concludes that the licensee has met the intent of GL 88-20 for conduct of an IPE to identify and understand the contribution of human performance to plant specific severe accident vulnerabilities.

5. Licensee Actions and Commitments

As part of the IPE process and in response to staff questions for clarification, the licensee identified the following hardware and procedural enhancements which were to have been implemented by the end of the 1993 refueling outage. However, as stated in the licensee's letter dated July 14, 1994, these changes have not been implemented.

1. The IPE has taken credit for a design modification which will allow the stand-by gas treatment filters to be isolated with valves rather than requiring operations and maintenance personnel to remove expansion joints and install blind flanges. This modification was to be installed during the 1993 refueling outage.

2. The EOP procedure associated with aligning containment venting was to be revised to add guidance on locally opening the outside containment purge valve when instrument air or division I emergency AC is unavailable, in addition to guidance on aligning instrument air to the nitrogen supply allowing the operator to open the inside containment purge valve when nitrogen is unavailable.

3. Guidance was to be added to procedures on opening doors from the auxiliary building into the pump rooms upon loss of cooling to HPCS, RCIC, and LPI pump rooms. NMPC believes that the flow of air from the pump room through a pipe chase back to the auxiliary building will protect the pumps from this event.

4. Additional guidance on opening doors and isolation of a flood source was to be provided to the operators for service or fire water system floods in an emergency diesel room or the control building. NMPC considers this important since all emergency AC is located in the area.

5. Precautions were to be added to the low pressure injection test and maintenance procedures to ensure that opening of the low pressure injection paths during power operation is unlikely.

6. As a result of the station blackout rule, NMPC has committed to develop station blackout specific emergency operating procedures.

   The IPE has taken credit for these procedures, which are to address:

- Bypassing the RCIC interlock circuitry within two hours which will prevent automatic high room temperature isolation and turbine exhaust backpressure trips.
- Shedding of all nonessential DC loads within the first 2 hours.
- Remote operability of RHR injection MOVs without AC power, to allow diesel fire pump injection.
- Instructions on operation of SRVs to minimize depletion of nitrogen and DC power, prevention of RPV pressure isolation of RCIC, or if RCIC fails, to depressurize to allow to diesel fire water pump injection.
- Explicitly include local closure of outside containment valves dependent on AC power.

In addition to the above, NMPC has identified additional insights (IPE section 6.3) which could result in system or procedural changes that may be considered in the future. However, the IPE identified the following for station blackout:

- "There is uncertainty about the capability of the diesel fire water to provide successful injection to the reactor pressure vessel through the RHR system by way of 100 feet of 2.5 inch hose. Test data and additional information or tests are being pursued to establish a system injection flow profile."

- "The HPCS would become another recovery option if fire water could be used to cool the HPCS diesel. There may be a relatively inexpensive modification that would provide this capability. Use of existing service water piping and/or unit 1 connections are being considered."

III. CONCLUSIONS

The staff finds the licensee's IPE submittal for internal events including internal flooding is consistent with the information requested in NUREG-1335. Based on the review of the submittal, the licensee's response to questions and associated information, the staff finds the licensee's IPE conclusion that no fundamental weakness or severe accident vulnerabilities exist at NMP-2 to be reasonable. The staff notes that:

(1) NMP-2 personnel were involved in the development and application of PSA techniques to the NMP-2 facility, and that the associated walkdowns and documentation reviews constituted a viable process for confirming that the IPE represents the as-built, as-operated plant.

(2) The licensee performed an in-house peer review to ensure that the IPE analytic techniques had been correctly applied and documentation is accurate.

(3) The front-end IPE analysis is complete with respect to the level of detail requested in NUREG-1335. In addition, the analytical techniques were found to be consistent with other NRC reviewed and accepted PSAs.

(4) The back-end analysis addressed the most important severe accident phenomena normally associated with the Mark II containment type. No obvious or significant problems or errors were identified.

(5) The HRA allowed the licensee to develop a quantitative understanding of the contribution of human errors to CDF and containment failure probabilities.

(6) The employed analytical techniques in the front-end analysis, the back-end analysis, and the HRA are capable of identifying potential plant-specific vulnerabilities.

(7) The licensee's IPE process searched for DHR vulnerabilities consistent with the USI A-45 (Decay Heat Removal Reliability) resolution.

(8) The licensee responded to CPI Program recommendations, which include searching for vulnerabilities associated with containment performance during severe accidents.

Based on the above findings, the staff concludes that the licensee demonstrated an overall appreciation of severe accidents, has an understanding of the most likely severe accident sequences that could occur at the NMP-2 facility, has gained a quantitative understanding of core damage and fission product release, and responded appropriately to safety improvement opportunities identified during the process. The staff, therefore, finds the NMP-2 IPE process acceptable in meeting the intent of GL 88-20. The staff also notes that the licensee has indicated that "as additional information and technology becomes available the IPE will be extended, updated and used by NMPC, and that based on the IPE and its updates, improvement initiatives will continue and the IPE as a living program, will continue to benefit the plant."

APPENDIX
NMP-2 DATA SUMMARY SHEET*
(INTERNAL EVENTS)

- o Total core damage frequency (CDF): 3.1E-5/Year

- o Major initiating events:

	Contribution (%)
Loss of offsite power	26
(Blackout	17)
(Non-blackout	9)
Loss of Emergency AC Division II	16
Loss of Emergency AC Division I	15
Loss of 115 kv Offsite Source A	8
Loss of 115 kv Offsite Source B	7
Loss of Condenser	3
Flood in EDG room Unisolated	3
Others	22

- o Major contributions by functional group:

	Contribution (%)
Loss of Injection (non-SBO)	50
Loss of Heat Removal	29
Station blackout (SBO)	18
ATWS	4
Floods	5

- o Major contributions to dominant core damage sequences:

  - Loss of all injection precipitated by loss of offsite power, followed by a) SBO due to independent failure of Division I and II AC power, RCIC, and failure to recover; or b) independent failure of HPCS, RCIC, and operator depressurization.

  - Loss of all injection due to loss of one division of emergency AC power and subsequent independent failure of the opposite train of DC power causing loss of service water and RCIC.

  - Loss of heat removal due to loss of one division of emergency AC power and subsequent independent failure of service water and consequential failures causing loss of RHR and containment venting leading to containment failure and injection failure.

- Major operator action failures (percentage importance measure, from response to staff questions):

  - Failure to initiate RPV depressurization (9 percent)
  - Failure to restore service water given loss of one offsite power source (7 percent)
  - Failure to open door to auxiliary bay room to establish room cooling given service water failure (7 percent)
  - Failure to establish containment venting given loss of air (5 percent)
  - Operator failure to align RHR for containment heat removal (3 percent)

- Conditional containment failure probability given core damage:

No Containment Failures	27 percent
Vented	04 percent
Failed (nonvented)	69 percent

- Significant IPE findings:

  - NMP-2 has three emergency diesel generators, the third of which is dedicated to the HPCS. HPCS system depends on the service water A and B, which are normally cross-tied, for diesel cooling and diesel room and HPCS pump room cooling. Consequently when both divisions of emergency AC power are lost, or one division and opposite division of DC power is lost, or one division (AC or DC) and the opposite train of service water fails independently, HPCS is lost.

  - As identified in the dominant sequences, loss of a single division of AC power subsequently causes all service water pumps to trip. The TBCLC and RBCLC system service water loads will be isolated on loss of the single division of power causing a low flow trip of the pumps in the opposite train of service water. This requires them to restart thus presenting additional failure modes for the service water system, complicating these events.

  - Support system initiating events are important contributors (greater than 75 percent) to the CDF, dominated by loss of a single division of either emergency AC, or offsite (115 kv) power (46 percent) and LOOP (26 percent).

  - 53 percent of the CDF ends in a high [early (3 percent), intermediate (37 percent), late (13 percent)] release category. Of the 37 percent of CDF ending in a high intermediate release, 19 percent are associated with accident class IA (loss of high pressure injection and failure of RPV depressurization), and 81 percent with class ID (loss of makeup at low RPV pressure). The majority of the sequences leading to these accident classes originate from loss of offsite power or loss of one division of emergency AC or offsite power initiating events.

- Important plant hardware (importance (percent) of the top events to CDF, excluding the impact of failure due to support systems or initiating events.)

  Emergency AC power [top events A2 (26 percent), A1 (23 percent)]
  RHR [top events LA (17 percent), LB (14 percent)]
  Containment Venting [top event CV (17 percent)]
  HPCS [top event HS (15 percent)]
  Service Water [top events SA (14 percent), SB (12 percent)
  RCIC [top events IC (12 percent), U1 (9 percent), U2 (3 percent)]
  Emergency DC power [top events DA (12 percent), DB (12 percent)]

- Enhanced procedures, hardware, and operator actions:

  - Modification of standby gas treatment system and procedures for containment venting

  - Procedure for auxiliary bay pump room cooling

  - Station blackout procedures

  - Internal flood analysis and procedural guidance

  - Procedural precautions for ISLOCA test and maintenance

(*Information has been taken from the NMP-2 IPE and the NMPC response to staff questions and has not been validated by the NRC staff.)

ENCLOSURE 2

NINE MILE POINT NUCLEAR STATION, UNIT 2 INDIVIDUAL PLANT EXAMINATION

TECHNICAL EVALUATION REPORT

(FRONT-END)