SEA 92-553-07-A:1

Revision 1

# Nine Mile Point 2 IPE: Front-End Audit

## Contractor Step 1 Audit Report

NRC-04-91-066, Task 7

F. Sciacca
J. Jessen
R. Walsh
R. A. Clark
J. Darby

Science and Engineering Associates, Inc.

Prepared for the
Nuclear Regulatory Commission

Table of Contents


| Section | | | Page |
|---|---|---|---|
| I. | INTRODUCTION | | 1 |
| | I.1 | SEA Audit Process | 1 |
| | I.1.1 | Review of FSAR and Tech Specs | 1 |
| | I.1.2 | Review of IPE Submittal | 2 |
| | I.2 | Nine Mile Point 2 IPE Methodology | 2 |
| | I.3 | Nine Mile Point 2 Plant | 2 |
| | I.3.1 | Similar Plants and PSA's | 3 |
| | I.3.2 | Unique Features | 3 |
| II. | REVIEW FINDINGS | | 5 |
| | II.1.1 | General Overview of Front-End Analysis | 5 |
| | II.1.1.1 | Completeness of Submittal with Respect to the Type of Information and Level of Detail Requested in NUREG-1335 | 6 |
| | II.1.1.2 | Description and Justification for Methodology Used, and Consistency with Generic Letter 88-20 Methods | 6 |
| | II.1.1.3 | Assurance of Use of As-Built, As-Operated Plant | 7 |
| | II.1.1.4 | Internal Flooding Methodology | 9 |
| | II.1.1.5 | Peer Review | 11 |
| | II.1.2 | Review of Accident Sequence Delineation and System Analysis | 11 |
| | II.1.2.1 | Identification of Initiating Events and Related Dependencies | 11 |
| | II.1.2.2 | Identification and Analysis of Front-Line and Support Systems Important to the Prevention of Core Damage and Mitigation of Fission Product Release | 13 |
| | II.1.2.3 | Treatment of Dependencies (Including Asymmetries) Among Plant Systems; Dependency Matrices | 16 |
| | II.1.2.4 | Treatment of Common Cause Failures | 17 |
| | II.1.2.5 | System Event Trees and Special Event Trees; Treatment of Initiating Events, Associated Success Criteria, and Dependencies Between Top Events | 18 |
| | II.1.2.6 | Identification of Most Probable Core Damage Sequences and Dominant Contributors; Consistency with Insights from PSAs of Similar Design | 23 |
| | II.1.2.7 | Treatment of Front-End and Back-End Dependencies: Sequences Screening Criteria, Consideration of Containment By-Pass, Consideration of Containment Isolation, Consideration of Reactor System/Containment System Availability in Plant Damage States, Source Term, System Mission Times, Inventory Depletion, Dual Usage (Spray vs. | |

Injection) 28
II.1.2.8 Consideration of Initiating Events Affecting More Than One Unit; Treatment of Systems Shared Between Units 30

II.1.3 Review of the IPE's Quantitative Process 31
II.1.3.1 Quantitative Evaluation of the Impact of Integrated System and Component Failures on Plant Safety; Use of Mean Values and Sensitivity Studies 31
II.1.3.2 Consistency of Techniques Used to Perform Data Analysis 33
II.1.3.3 Sources of Generic Failure Data and Rationale for their Use 33
II.1.3.4 Common Cause Failure Data and Data Sources 35

II.1.4 IPE Approach to Reducing the Probability of Core Damage or Fission Product Release 36
II.1.4.1 Core Damage Vulnerability and Efforts to Uncover Vulnerabilities; Plant Modifications (or Safety Enhancements) to Eliminate or Reduce the Affect of Vulnerabilities 36
II.1.4.2 Identification of Plant Improvements and Proposed Modifications Expected to Enhance Plant Safety 37

II.1.5 Licensee's Evaluation of the Decay Heat Removal Function 39
II.1.5.1 Reliability of the DHR Function and Consistency With Other PSA Findings 39
II.1.5.2 IPE Consideration of Diverse Means of Decay Heat Removal 40
II.1.5.3 Decay Heat Removal Unique Features 41
II 1.5.4 Conclusion Regarding NMP2 Evaluation Decay Heat Removal 41

III. OVERALL EVALUATION AND CONCLUSION 42

IPE REVIEW
NINE MILE POINT UNIT 2

# I. INTRODUCTION

This introduction presents the process used by Science and Engineering Associates (SEA) to review the front end portion of the Niagara Mohawk Power Corporation (NMPC) Individual Plant Examination (IPE) Submittal for the Nine Mile Point Nuclear Station Unit 2 (NMP2). This front end review focuses on accident sequences leading to core damage, due to internal initiating events and internal flooding. Reviews of the human factors and back end aspects of the Nine Mile Point 2 IPE were performed by the NRC with contractual assistance from Concord Associates, Inc., and Scientech, Inc., respectively.

## I.1 SEA Review Process

The evaluations presented herein are the results of a "submittal only" review. As such, the review addressed the NMP2 IPE submittal documentation. The purpose of this review is to identify areas which may need to be investigated further with respect to the licensee meeting the intent of Generic Letter 88-20.

### I.1.1 Review of FSAR and Tech Specs

The NRC provided the Nine Mile Point 2 IPE Submittal to SEA in September, 1992. SEA began work on the Nine Mile Point 2 review in late September, 1992.

In October, 1992 selected portions of the latest (updated) Safety Analysis Report (USAR) and Technical Specifications (Tech Specs) for Nine Mile Point 2 were copied and made available to SEA's lead analyst. These copies were made from up-to-date documentation provided by the NRR Project Manager.

### I.1.2 Review of IPE Submittal

A detailed review of the IPE Submittal for Nine Mile Point 2 was accomplished. The effort incorporated a complete review of all aspects of the front end identified in the Statement of Work (SOW) for this task and NUREG-1335. In addition, the guidance provided in the "Draft Step 1" Review Guidance Document," dated May 19, 1992, was utilized.

## I.2 Nine Mile Point 2 IPE Methodology

The Nine Mile Point 2 IPE uses the large event tree/small fault tree methodology to perform the front end analyses. A support system event tree was used to evaluate support systems and their impacts on front-line systems. Recovery actions are considered. Common Mode failures are incorporated into the fault tree models. Importance analyses were performed on selected parameters, which included functionally grouped sequences (e.g., ATWS, station blackout, loss of heat removal, loss of injection, and internal floods), initiating events, event tree top events, split fractions, contributors to split fractions, and human actions. .

The methodology used in the IPE front end analysis of NMP 2 is consistent with the methodology identified in NUREG-1335.

## I.3 Nine Mile Point 2 Plant

Nine Mile Point Unit 2 is a General Electric designed Boiling Water Reactor (BWR) employing a BWR 5 nuclear steam supply system (NSSS). The rated thermal power is 3323 MWt, and the nominal power output rating is 1080 MWe. NMP2 has a Mark II type containment which utilizes an over-under pressure suppression design with multiple downcomers connecting the drywell to the suppression pool.

The Nine Mile Point Nuclear Station Unit 2 is located on the southeast shore of Lake Ontario. It is located about 6 miles northeast of the city of Oswego, New York. The Nuclear Steam

System Supplier was General Electric Co., and Stone and Webster was the Architect Engineer and Constructor. The NMP2 project was granted a construction permit in 1974, and the unit achieved commercial operation in 1988.

### I.3.1 Similar Plants and PSA's

The following plants are General Electric BWR 5 designs [NUREG/CR-5640]:

- LaSalle 1 and 2
- WNP-2

General Electric BWRs with the Mark II type containment designs include:

- LaSalle 1 and 2
- WNP-2
- Susquehanna 1 & 2

Probabilistic Safety Studies (PSA) for BWR plants similar to Nine Mile Point 2 include those performed for Peach Bottom Unit 2, LaSalle, and Grand Gulf Unit 1.

:on

### I.3.2 Unique Features

The Nine Mile Point Unit 2 submittal identified several "interesting" design features during the IPE evaluation process. The features of interest to the Level 1 evaluations are as follows:

ATWS. The redundant reactivity control system (RRCS) at NMP2 automatically actuates standby liquid control (SLC), reactor recirculation pump trip, alternate rod insertion, and feedwater runback. This system was assessed to be reliable and negated the need to model operator actions associated with these functions. Other operator actions associated with level control are not dependent on manual initiation of SLC or the other functions.

Spatial Considerations. The separation of the auxiliary bays, submarine type doors to the auxiliary by pump rooms, high pressure core spray system (HPCS), and reactor core isolation cooling system (RCIC) provide substantial protection from floods and other hazards. On the other hand, this spatial protection provides difficulties for equipment when room cooling is lost. However, redundancy in room cooling units provides reliable cooling in comparison to a single pump. It takes a total loss of service water to require recovery actions associated with opening doors and protecting pumps. Several air conditioning systems in the control building provides significant redundancy with regard to opening doors to other areas with separate air conditioning.

HPCS. The HPCS system is independent with respect to actuation system inputs and emergency AC (referred to as Division III). However, during a station blackout the HPCS is unavailable because the HPCS diesel depends on service water which is unavailable during a station blackout since power for service water is provided by the other diesel generator. A potential improvement discussed below has been identified for consideration which would allow a HPCS success path.

Interfacing Systems LOCA. The frequency of an interfacing systems LOCA was assessed to be low frequency at NMP2 due primarily to extra strength pipe used for piping diameters greater than 12 inches. Piping less than or equal to 12 inches is standard or extra strength pipe. The RHR shutdown cooling suction path has power removed from the motor operated valves during power operation and there is a third normally-closed motor operated valve in each pump suction paths. The low pressure injection paths are not stroke tested during power operation and procedural precautions are being added to testing and maintenance procedures to reduce the likelihood of inadvertent opening.

Offsite Power Connections. The 345 kV and 115 kV connections are physically separate outside the plant located in the switchyard. There are cross-tie capabilities inside the plant which allow one 115 kV source to supply all divisions of emergency AC power.

DC Power. The DC power system is divided between a non-safety related subsystem and a safety related subsystem. The safety related subsystem is completely independent of the nonsafety subsystem. This greatly improves DC reliability as load-shedding of numerous non-safety loads is not necessary to protect the safety related loads. Some DC load shedding is warranted, but because of the separation, it is limited to a relatively few loads.

Hardened Containment Vent Capability. The availability of the containment vent system and procedures give NMP2 operators an additional set of mitigation actions to take in an emergency.

Motor Driven Feedwater Pumps. With the exception of support system failure; feedwater (injection) and condenser (heat removal) were determined to be independent. That is, loss of condenser initiating events do not cause loss of feedwater at NMP2.

Containment Flooding Capability. Containment flooding systems and procedures are available which give NMP2 personnel an additional set of mitigation actions to take in an emergency.

## II. REVIEW FINDINGS

This section discusses the Nine Mile Point 2 Station IPE review findings. The organization strictly follows that specified under subtask 1 of the Statement of Work for this task. Topic headings are in bold type, as are summary statements and additional information needs regarding particular topics.

### II.1.1 General Overview of Front-End Analysis

#### II.1.1.1 Completeness of Submittal

The Nine Mile Point 2 IPE submittal was, with minor exceptions, organized and presented strictly according to Table 2-1 "Standard Table of Contents for Utility Submittal" provided in NUREG-1335. The only exceptions were that Section 3.2.2 "System Analysis" from NUREG-1335 was not used, but the System Analysis is presented in Section 3.2.1. Also, Sections 4.8 and 4.9 of the submittal discuss "Accident Management Insights" and "Sensitivity Analysis", respectively. There were no obvious omissions. The submittal contains the type and level of detail requested in NUREG-1335.

### II.1.1.2 Description and Justification for Methodology Used

The Nine Mile Point 2 IPE documentation indicated that the methodology followed was similar to that of NUREG/CR-2300. This IPE used the large event tree/small fault tree method. Support system event trees were used. The methodology used is briefly summarized in Section 2.3 of the submittal, and is more thoroughly discussed in appropriate locations in the documentation. The stated attributes of the selected approach are that it allows for an explicit accounting of the performanco of each system in a sequence, that it enhances accounting for support system effects, and that it automated the binning of like sequences. The chosen approach also directly linked the event trees in such a way that each sequence is tracked from initiating event to the Level II end state.

In the identification of initiating events, the Nine Mile Point 2 submittal indicates that other PRA's for similar plants were reviewed, as were BWR generic experience data. Nine Mile Point specific operating experience was also reviewed to identify initiating events. All major plant systems were subjected to an FMEA to further assist in initiating event identification.

Event sequence diagrams were used to help document required system and operator response to initiating events and to assist in the plant modeling through detailed event trees. System interdependencies were modeled in the event trees, as were human interactions. Systemic event trees were used, and the tree top events represent system responses and operator actions.

Although several event trees are presented, for each initiating event the linking of the event tree in essence created one very large event tree during quantification. Thus, binning of the Level 1 results was not necessary prior to performing the Level 2 evaluations. This approach resulted in a fully integrated front-end/back-end interface.

The dependencies among systems were generally well documented and appear to have been carefully considered.

The system descriptions provided included tables which listed the components included in the fault tree model for each system. Thus, the fault trees appear to have been developed down to the component level. The fault tree logic models are stated to have been quantified using the RISKMAN software. Quantitative results, in terms of system failure rates under various conditions, were also presented (Table 3.3.5-1).

**In summary, the methods used to perform the Nine Mile Point 2 IPE are clearly described and their basis for selection, as discussed above, is reasonable. The methodology employed is the large event tree/small fault tree approach. It is judged to be consistent with the methods identified in Generic Letter 88-20.**

### II.1.1.3 Assurance of Use of As-Built, As-Operated Plant

Section 2.2 states that the Nine Mile Point 2 IPE was based on the plant configuration as it existed at the completion of the first operating cycle, and subsequent documentation provided by the licensee indicates the "freeze date" for the IPE as February, 1990. Three levels of plant walkdowns were also employed in the evaluation process. These included a walkdown of primary containment and the containment contained therein during plant shutdown, one focusing on the reactor building structure, and the third level consisting of numerous individual walkdowns as necessary for individual systems analyses and internal flood evaluations. The submittal does not indicate whether or not the walkdowns identified discrepancies in actual system/component configuration or location compared to the plant drawings or other documentation. The walkdowns were stated to be important to "supplement plant information, verify plant information, and give the analysts an appreciation for plant operation".

Section 6 of the submittal indicated that the model used to perform the IPE analysis did take credit for certain physical modifications and procedural improvements which had not been fully implemented at the time the IPE was submitted to the NRC. The physical modifications are scheduled to be made during the refueling outage scheduled for December 1993. The procedural changes are stated to be under development and review. The quantitative impact of these improvements on core damage frequency or on IPE conclusions was not discussed in the submittal. Subsequent licensee responses to NRC questions indicated that the analysis took credit for five enhanced operator actions. These procedural changes were stated to improve operator reliability for performing important actions during accidents. The impacts were not quantified.

Many sources of plant documentation were stated to have been used in establishing the plant model. This included the Updated Safety Analysis Report (USAR), plant procedures (plant operation, emergency operating, maintenance, surveillance, special operating), plant drawings, design basis documents (calculations, etc.), equipment qualification reports, licensee event reports, inservice test results, and maintenance work reports. The origin or latest version dates of the documentation used were not specified.

**In summary, the IPE documentation indicates that the analysis performed reflected the plant design as of February 1990, and that plant walkdowns and other methods were employed to help assure that this was the case. The models employed took credit for certain procedural changes which have not yet been implemented, but which are planned to be implemented. The impact of these modifications and procedural changes on CDF was not discussed quantitatively.**

### II.1.1.4 Internal Flooding Methodology

The NMP flooding analysis includes flooding of equipment by large volumes of water (i.e., equipment submergence). No consideration was identified in the submittal for water spraying, dripping, or splashing of water on sensitive equipment. However, in response to staff questions the licensee indicated that water intrusion from spray or direct impingement, such as splashing, were included in the assessing risks associated with flooding events.

The NMP flooding methodology includes the following steps:

1. Plant Familiarization - This step was performed to familiarize the analysts with the location of potential flood sources and the pathways available for the propagation of a flood.
2. Flood Experience Review - Industry data (Nuclear Power Experience) concerning actual occurrences of flooding at nuclear power facilities were reviewed to ensure familiarization with actual flooding scenarios; specifically, their causes and effects.
3. Flood and Equipment Locations - Using the information from Steps 1 and 2, potential flooding scenarios were postulated.
4. Plant Walkdown - A plant walkdown was performed to collect additional information and to confirm previous documentation, flood sources, and impacts.
5. Scenarios and Screening- Using conservative assumptions about flood size and system impacts, the scenarios from Step 3 are initially screened.
6. Quantification - The flooding scenarios not screened out in Step 5 were further analyzed and quantified to obtain core damage frequencies.

Critical Internal Flood Areas Identified:

- Emergency Diesel Generator Rooms
- Control Building El. 261'

- Turbine Building El. 250'
- Service Water Pump Bay

Most Critical Flood Sources Identified:

- Service Water System
- Circulating Water System
- Firewater System

Most Critical Flood Scenarios Identified:

- Flooding of the Control Building El. 261' by the Fire Water System
- Flooding of the Emergency Diesel Generator Rooms by the Service Water System
- Flooding of the Turbine Building El. 250' by the Service Water System
- Flooding of the Service Water Pump Bay by the Service Water System
- Flooding of the Turbine Building El. 250' by the Circulating Water System

The contribution to core damage from internal flooding events was calculated to be 1.55E-06/yr or 5% of the total core damage frequency. This contribution is dominated by a service water leak originating in an emergency diesel generator room (Control Bld 261'). The leak is not isolated and eventually floods the emergency switchgear rooms causing an unrecoverable station blackout and core damage.

It is our judgment that the methodology used in the NMP flooding analysis is sufficient to identify the dominant flooding core damage sequences.

### II.1.1.5 Peer Review

Section 5 of the Nine Mile Point 2 IPE submittal describes the project review process followed by Niagara Mohawk Power Corporation (NMPC). The Quality Assurance Department (QA) and the Independent Safety Engineering Group (ISEG) were assigned responsibility for the independent in-house review. Twenty-two individuals are listed as having participated in the review process, only one of which in identified as not being an employee of NMPC. These individuals represented the disciplines or areas of expertise of quality assurance, system engineering, training, nuclear technology, design, electrical design, mechanical design, structural design, operations, plant evaluation, emergency preparedness, and licensing. The reviews were stated to take place on essentially a continuous basis. The main results of the reviews are stated to be a better understanding of the plant operation and design. The reviews also improved the accuracy of the IPE relative to the plant as designed and operated by NMPC.

**Based on the descriptions provided, it is concluded that the Nine Mile Point 2 IPE models and analysis were subjected to reviews by plant staff familiar with the plant systems and plant operation.**

### II.1.2 Review of Accident Sequence Delineation and System Analysis

#### II.1.2.1 Identification of Initiating Events and Related Dependencies

The IPE submittal states that the impact of each initiating event on systems that might be used for mitigation was identified.

This review examines the process used to identify initiators which the licensee referred to as a comprehensive engineering evaluation. Internal flood initiators and dependencies are included in the review.

The IPE submittal states that the identification of initiating events was based on a review of generic sources of information and plant-specific evaluations. The generic sources included:

- Surveys of industry data through 1983 (IPE Refs. 2 and 3);
- A supplementary LER search (1984 through 1990); and

- PRAs for Brunswick (IPE Ref. 4), Shoreham (IPE Ref. 5), Limerick (IPE Ref. 6), Peach Bottom (IPE Ref. 7), Grand Gulf (IPE Ref. 8), and WASH-1400 (IPE Ref. 9).

The transient initiator categories used for the IPE were consolidations of the BWR transient initiating events listed in NUREG/CR-2300. Table 3.1.1-3 of the submittal shows how the transient initiator categories were consolidated. The licensee states that the groups of transients were determined to impact plant systems, potential recovery actions, and plant conditions in a similar manner.

The submittal provides a list of plant-specific support system initiators which were generated by considering the plant impact of failures of major systems, including whether the failure causes a plant shutdown or trip. The underlying evaluation is the system-level FMEA, which the submittal documents.

The initiating events are consistent with the NRC-sponsored Peach Bottom PRA (NUREG/CR-4550, Vol. 4); the small LOCA in the IPE includes both the small and small-small LOCAs of the Peach Bottom PRA.

The NMP2 evaluation of LOCA initiators in Section 3.1.1.3.2 discussed LOCAs outside containment. High pressure piping breaks (feedwater and main steam) were stated to be considered a negligible contribution to risk in comparison to other LOCAs. These initiators were excluded from the analysis on the basis that the frequency of rupture of these lines is very much less than for lines inside containment. Therefore, they were qualitatively screened from the analysis.

The licensee provided a list of mitigating functions required for each initiator and provided dependency tables that identify dependencies between systems.

As described in the IPE submittal, the process used to identify internal flood initiators and potential dependencies consists of :

- review of details of the plant layout to familiarize analysts with the location of potential flood sources and pathways available for propagation,
- review of causes and effects of actual flooding at nuclear power facilities from Nuclear Power Experience (IPE Ref. 39), and
- plant walkdown.

The submittal does not explicitly state that flood initiators such as tank overfilling or pump seal leaks were considered; however, we assume that any significant events caused by such initiators were captured in the nuclear plant experience base reviewed. The flooding initiator events are discussed in qualitative rather than quantitative terms. As such, the licensee did not indicate the flow rates or other assumptions that affect the time available for mitigation.

**Considering also the information provided in Sections 2 and 3.1.1 of the NMP2 IPE submittal, we conclude that the licensee has described the process used to identify initiators, including internal flood, and that the IPE takes into account both generic and plant-specific information. Furthermore, the initiating events appear to be consistent with those considered in other PRAs.**

### II.1.2.2 Identification and Analysis of Front-Line and Support Systems Important to the Prevention of Core Damage and Mitigation of Fission Product Release

The systems modeled in the Nine Mile Pt 2 IPE are listed below. The support systems included:

- Normal AC Electrical Power
- Emergency AC Electrical Power, Div. I, II, and III
- Normal DC Electrical Power
- 125V DC Electrical Power, Div. I, II, and III
- 120V AC, Uninterruptible Power Supplies
- Reactor Recirculation System, Div. I and II
- Emergency Core Cooling System (ECCS) Actuation Signal System, Division I and II

- Instrument Air
- Nitrogen System
- Service Water System
- Reactor Building Closed Loop Cooling System
- Turbine Building Closed Loop Cooling System
- Heating, Ventilation, and Air Conditioning (HVAC)*

The front-line systems included in the IPE model are listed below. The listing includes containment systems.

- Reactor Core Isolation Cooling
- High Pressure Core Spray
- Steam Relief Valves/Automatic Depressurization System
- Low Pressure Core Spray
- Residual Heat Removal System, Trains A, B, and C
- Control Rod Drives
- Redundant Reactivity Control
- Fire Water Crossties to RHR
- Condensate
- Feedwater
- Reactor Protection System
- Containment Venting System
- Containment Isolation System
- Vapor Suppression*
- Alternate Rod Insertion
- Standby Liquid Control
- Main Steam Isolation Valves
- Turbine Bypass
- Reactor Recirculation Pump Seal Integrity
- Drywell Cooling

- Vessel and Primary Containment Instrumentation*

The Nine Mile Point Unit 2 IPE submittal described each of the foregoing systems. The discussions for each system presented the following information:

- system function,
- success criteria for each particular system as applied to event tree specific top events,
- support systems required for the success of system of concern as related to event tree top events,
- system operation,
- instrumentation and control requirements,
- technical specifications related to the system,
- surveillance, testing and maintenance applicable to the system,
- initiating event potential (potential for system to cause an initiating event),
- equipment location,
- operating experience,
- modeling assumptions,
- logic model (statement of fault tree use),
- system schematics/drawings,
- event success diagrams (for certain systems),
- tables of "component block descriptions", and
- documentation references.

The component block descriptions listed major components of each system and subsystem, their failure modes, initial and actuated states, support systems relied upon by the components, and the results of loss of support systems.

The items flagged with an asterisk (*) were not explicitly included in the dependency matrices listed in Tables 3.2.3-1 and 3.2.3-2 of the documentation. However, they are discussed in the documents and the foregoing topics are covered for these systems as well as those included in the dependency matrices. Note that the HVAC was not included in the dependency matrices, but its important functions for other support systems and for frontline systems are included in the models of the systems being supported (Section 3.2.1.11).

**Overall, the Nine Mile Point Unit 2 IPE documentation indicates that the preparers performed a thorough review of the frontline and support systems important to the prevention of core damage and mitigation of fission product releases.**

### II.1.2.3 Treatment of Dependencies (Including Asymmetries) Among Plant Systems; Dependency Matrices

Section 3.2.3 of the Nine Mile Point 2 IPE documentation specifically addresses dependencies. This type of information is also presented in the individual system descriptions (Section 3.2.1) and, to a lessor degree, in the discussion of initiating events (Section 3.1.1). Table 3.2.3-1 presents support-to-support system dependencies, and Table 3.2.3-2 presents support to frontline system dependencies. Each dependency indicated in the tables is explained or briefly discussed so that the nature and specific elements of the dependence is provided.

The reliance on electric power, ESF actuation , instrument air, service water, and the reactor and turbine building closed loop cooling systems are explicitly noted in the dependency matrices. However, equipment dependencies on HVAC are not included in Tables 3.2.3-1 and 3.2.3-2, but are discussed in the system descriptions. As noted previously, system dependencies on HVAC were included in the model of each such dependent system.

Asymmetries within systems in the plant appear to be accounted for in the evaluation process. For example, Nine Mile Point Unit 2 employs three residual heat removal (RHR) trains, designated A, B, and C. The trains are not identical, nor do they all perform the same functions. Trains A and B can provide suppression pool cooling, whereas train C cannot. The descriptions and evaluations provided in the documentation indicate that such asymmetries were accounted for.

**Overall, the review indicates that the licensee had adequately considered dependencies and asymmetries between and among systems. No deficiencies were identified.**

### II.1.2.4 Treatment of Common Cause Failures

The IPE modeled those common cause failure modes within each system that satisfied the following screening criteria presented in NUREG/CR-4780 (IPE Ref. 41):

- components are identical,
- components represent redundancy in the failure logic model,
- components are active rather than passive, and
- the contribution of common cause failure is not expected to be insignificant relative to the contribution of other failures, including other common cause failures.

The IPE used the Multiple Greek Letter (MGL) method to quantify common cause failures within a system.

No plant-specific common cause component data were used because NMP2 is a relatively new plant (beginning operation in 1988) with very little common cause failure experience. Therefore, there was no plant-specific common cause failure data to examine for root causes.

**The IPE used the MGL method to model common-cause failures within system, using generic data. Root-cause analysis of plant-specific data would have been premature given the youth of the plant.**

### II.1.2.5 System Event Trees and Special Event Trees; Treatment of Initiating Events, Associated Success Criteria, and Dependencies Between Top Events

The Nine Mile Point 2 IPE employed the large event tree, small fault tree approach. Both frontline and support system event trees were utilized. All trees are systemic and the top events (nodes) address success or failure of key systems to operate. Necessary operator actions are also included in the tree nodes.

**Front-Line Event Trees**

Frontline event trees were developed for Nine Mile Point 2 responses to transients, station blackout, anticipated transient without scram (ATWS), small LOCA, medium LOCA, and large LOCA events. Each response model is made up of two frontline event trees, the first for the earlier or initial response activities and the second for subsequent actions.

Success Criteria. Success criteria were presented and discussed at several levels (Section 3.1.1, Tables 3.1.1-7 through 3.1.1-9). Overall success criteria were discussed in terms of the achievement of adequate core cooling, maintaining containment integrity, maintaining reactor pressure vessel (RPV) integrity, and in terms of mission time. Functional success criteria were specified based on initiating events, and included specification of the systems or portions of systems needed to achieve reactivity control, reactor overpressure control, reactor system pressure control, inventory makeup (separately specified for high pressure, depressurization, and low pressure conditions), and containment pressure control. Operator actions necessary to achieve success were also specified, as were mission times. In addition, for each tree top event for each tree, the requirements for success were further discussed and defined.

The basis for the success criteria were stated to be based on deterministic analyses. Both BWR-generic and Nine Mile Point Unit 2 specific analyses were utilized. These analyses were used to help determine minimum requirements to prevent onset of core damage, maintain containment integrity, and maintain RPV integrity. The plant-specific analyses used the MAAP code.

Success criteria are stated to be based on realistic evaluations of critical parameters.

Event Identification. A check of the event trees indicates the event tree headings are in the appropriate order chronologically, and that they are consistent with specified success criteria.

The spot checks also indicated that the event tree headings were consistent with dependency matrix information. Event Sequence Diagrams (ESDs) were also provided in the submittal. The ESDs graphically display the flow of events, required actions, and alternative actions available to achieve successful outcomes for the various accident sequences. The ESDs also indicate the key operator actions and system or component operation needed to achieve success at given event tree nodes. Thus they indicate the key elements making up the model for the event tree nodes.

Dependency Identification. The dependence of event tree top nodes or events on the outcomes of previous nodes are addressed in the submittal discussions of each tree. Also, the split fraction logic is provided in the document. This logic identifies key dependencies determining the split fractions. Assumptions used in assessing the success or failure at each branch point in the trees are discussed. Overall, the NMP2 IPE submittal demonstrates a careful consideration of dependencies.

Sequence Transfers. The Nine Mile Point 2 IPE submittal states that all sequences are automatically transferred through each tree pertinent to that sequence, and that binning is used only as an aid to understanding the Level 1 analysis. All initiating events pass through the support system event tree and are linked with the appropriate front-end event trees. At the end of the frontline event trees the sequences are binned to either success or core damage. However, as described by the licensee the binning is a convenience used in the assessment of Level 1 and to provide a useful method of discussing groups of sequences that potentially impact the Level 2 analyses (e.g., deterministic evaluations). Sequences that require further evaluation in the Level 2 analysis are transferred to the containment event trees. Logic rules are specified in terms of the top event successes and failures and the Level 2 event trees are linked directly to the Level 1 model such that sequences are quantified from initiating event to release category (Level 2 end states). Sequence transfers are handled automatically within the RISKMAN code used to perform the quantification.

Timing. Timing of operator actions, frontline system actuation and operational mission times as presented in the success criteria tables (Tables 3.1.1-7 through 3.1.1-9) and are discussed with the narratives provided regarding specific tree top events. The basis for the timing was discussed for certain systems and certain operator actions, but not for all. However, the success criteria tables do provide references as to the sources of the information used in the IPE evaluations.

**Notes on Specific Frontline Trees:**

Transient Event Tree. Success requires scram (power control), injection (level control), and containment heat removal (primary containment control). The tree structure was consistent with the ESD for transients, and the discussion of the nodes indicated sound consideration of complex interactions, operator actions, options available, etc. No deficiencies were identified.

Small LOCA Tree. The tree structure and the top event ordering were logically arranged. The events and structure reflect the functional success criteria for transients and small LOCAs provided in Table. With the exception of a top event questioning containment vapor suppression capabilities, the small LOCA tree is identical to the general transient tree. No deficiencies were identified.

Medium LOCA Tree. The tree structure and the top event ordering were logically arranged. The LOCA event tree success criteria of Table 3.1.1-9a indicated that depressurization was not required; however, the medium LOCA tree modeled this event. No deficiencies were identified.

Large LOCA Tree. The tree structure and the top event ordering were logically arranged. No deficiencies were identified.

**Special Event Trees**

ATWS. Comparisons of the Nine Mile Point 2 IPE results with those presented in NUREG-1150 for BWRs (Peach Bottom and Grand Gulf) indicate that the contributions of ATWS for NMP2 are within the range cited for the NUREG-1150 plants. The NMP2 ATWS contribution to mean core damage frequency (CDF) was given as 1.1e-6/yr, whereas those for Peach Bottom and Grand Gulf were estimated to be 1.9e-6/yr and 1.1e-7/yr, respectively. The treatment appears to be consistent with the NUREG-1150 treatment for this event.

Station Blackout. The station blackout tree was logically arranged to account for the different time phases and different recovery options available. The tree arrangement and supporting discussion clearly indicated that considerations such as battery depletion and recovery of AC power were accounted for in a reasonable manner. Reactor recirculation pump seal LOCA given loss of seal cooling (as from a station blackout) is discussed in Section 3.2.1.25 "Reactor Recirculation Pump Seal LOCA". That discussion indicates that the evaluations of the station blackout transient considered the impact of a seal LOCA on how long RCIC can operate. These evaluations indicated that seal LOCA has no impact on the success of RCIC, and therefore it was not modeled in this event tree. No deficiencies were identified.

Interfacing System LOCAs. Interfacing system LOCA pathways are discussed in Section 3.1.1.3.2.2, and the associated special event tree is discussed in Section 3.1.3.1. Potential interfacing system LOCA pathways were identified by reviewing all such pathways connected to the reactor primary coolant system. These were then screened using specified criteria to establish those which represented credible LOCA pathways. For the pathways that met the screening criteria, all valves in the pathways were considered. Isolation valve surveillance and maintenance were also discussed and taken into account in the evaluation. The treatment of interfacing system LOCAs appears to be thorough and sound.

**Support System Event Tree**

The NMP2 IPE utilized a single, very large support system event tree made up of 29 top events. The tree branched at every top event, and had in excess of 200 million possible outcomes. Thus, the tree structure shown in the submittal was completely general. The following systems and events were represented in the support system event tree.

Offsite AC Grid

115 kV Sources A and B

Recovery from a partial Loss of Offsite Power

Division I Battery

Division II Battery

Division I Emergency AC

Division II Emergency AC

Normal DC and AC from Source A

Normal DC and AC from Source B

Division I Emergency DC

Division II Emergency DC

Uninterruptible Power Supply - Source A

Uninterruptible Power Supply - Source B

ECCS Logic Division I

ECCS Logic Division II

Manual ECCS Actuation

Service Water, Train A

Service Water, Train B

Reactor Building Closed Loop Cooling Water

Turbine Building Closed Loop Cooling Water

North Aux. Bay MCC Room Cooling

South Aux. Bay MCC Room Cooling

Instrument Air

High Pressure Instrument Nitrogen (gaseous)

Instrument Nitrogen (liquid)

Condensate Storage Tank A

Condensate Storage Tank B

Note that HVAC as a support system is specifically listed only for the Auxiliary Bay MCC rooms. However, the IPE submittal states that important functions performed by HVAC systems were included in the other support or front-line systems. Specific HVAC dependencies were discussed in the respective system descriptions as well as in the discussions of equipment survivability.

The headings used in the support system event tree were consistent with the headings/systems listed in the dependency matrices.

Asymmetry Identification. Asymmetries were not explicitly discussed. However, the brief discussions provided of the support system event tree top events did point out or account for certain asymmetries. The dependency tables and split fraction tables also indicate consideration of asymmetries.

### II.1.2.6 Identification of Most Probable Core Damage Sequences and Dominant Contributors; Consistency with Insights from PSAs of Similar Design

The other PRA results available for comparison with the NMP2 IPE were the two BWRs included in the NUREG-1150 studies: Peach Bottom Unit 2 and Grand Gulf. The NMP2 IPE submittal also presented a comparison of results generated for a Shoreham nuclear plant PRA. The following tabulation presents a comparison of the overall Level 1 risk results.

| Accident Type | Nine Mile Point 2 | Peach Bottom 2 | Grand Gulf |
|---|---|---|---|
| Station Blackout | 5.5E-6 | 2.2E-6 | 3.9E-6 |
| ATWS | 1.1E-6 | 1.9E-6 | 1.1E-7 |
| LOCA | 8.0E-7 | 2.6E-7 | <1E-7 |
| Transients | 2.4E-5* | 1.4E-7 | <1E-7 |
| Total | 3.1E-5 | 4.5E-6 | 4.0E-6 |

*Includes small LOCAs, floods, and non-SBO loss-of-offsite power events

The estimated total core damage frequency (CDF) for NMP2 is about a factor of six higher than that estimated for Peach Bottom 2 or Grand Gulf. Both Peach Bottom 2 and Grand Gulf had

CDFs dominated by station blackout sequences. For NMP2, the largest contributor was transients dominated by loss of Divisional AC (emergency or offsite) power. Thus, all plants show sensitivity to loss of power type accidents. For all three reactors LOCAs appear to represent a relatively minor contributor to overall risk. Thus, the risk contributors for NMP2 appear to be generally consistent with those cited for other BWRs for which comparative data is available.

The dominant accident sequences listed in the NMP2 IPE submittal are presented in Section 1.4, "Summary of Major Findings." A listing of the top 100 CDF sequences is given in Section 3.4, "Results and Screening Process." The Section 1.4 discussions review the 10 accident sequences with the largest contribution to core damage. These top 10 sequences contribute about 40% to the overall CDF. For each sequence the IPE submittal lists the initiating event, subsequent failures of systems and/or components, consequential effects on other systems and/or components, and the annual frequency of the sequence (see Table 1.4-1). The failures of systems and components included failures of operators to take the proper action in the time available.

The discussions of dominant accident sequences presented the relative importance of systems regarding the prevention of core damage. The important systems include:

Emergency AC Power
Residual Heat Removal (RHR)
Containment Vent
High Pressure Core Spray (HPCS)
Service Water
Reactor Core Isolation Cooling (RCIC)
Emergency DC Power

The submittal also listed the following operator actions as being most important for the prevention of core damage:

AC Power Recovery
Containment Venting
Emergency Depressurization
Operation of Service Water
ECCS Pump Room Cooling.

The following paragraphs indicate the level of discussion and type of information presented for the top four accident sequences. The discussions indicate that the sequences were expanded to identify the dominant contributor(s).

**CDF Sequence 1 (2.6E-6/yr)**

The initiating event is a loss of offsite AC power (LOSP = 0.04/yr) which disables normal operating non-safety systems such as the condenser, feedwater, reactor building closed loop cooling (RBCLC), turbine building closed loop cooling (TBCLC), and instrument air. In addition, normally operating safety systems such as service water must restart on demand after the emergency diesels start and load. Given the LOSP initiating event, the following additional failures lead to core damage frequency:

- Division I emergency diesel generator fails or is in maintenance (A12 = 5.3E-2).
- Division II emergency diesel generator fails or is in maintenance (A28 = 6.5E-2).
- Division III emergency diesel and HPCS is guaranteed unavailable because service water is unavailable.
- Offsite AC power is not recovered within 30 minutes (I11 = 0.28).
- An emergency diesel (Division I or II) is not recovered within 30 minutes (G11 = 0.91).
- The operators successfully shed DC loads and disable RCIC trips within the first 2 hours to protect RCIC's availability. However, the RCIC fails either due to equipment failure or it is in maintenance (U11 = 7.6E-2). No credit is given to aligning the diesel fire pump within the first 2 hours of a station blackout because

of the time required to perform these actions and insufficient flow rates to the RPV.

**CDF Sequence 2 (2.4E-6/yr)**

The initiating event is a loss of Division II Emergency AC power (A2X = 4.3E-3/yr) which disables all safety systems that depend on Division II emergency AC. All Division II service water pump breakers open which causes isolation of RBCLC and TBCLC. Loss of cooling to the condenser, feedwater, and turbine generator equipment requires an immediate shutdown by the operators. Isolation of TBCLC and RBCLC is assumed to result is a low flow trip of the opposite Division service water pumps, thus requiring them to restart. In this sequence, the Division I battery fails (DA1 = 6.6E-4) on demand which prevents the restart of Division I service water pumps, fails RCIC, and prevents the start of Division I safety systems such as RPV injection. All service water is unavailable, feedwater is unavailable, HPCS fails due to loss of room cooling, and thus, all injection is unavailable.

**CDF Sequence 3 (2.3E-6/yr)**

The initiating event is a loss of Division I emergency AC power (A1X = 4.3E-3/yr) which disables all safety systems that depend on Division I emergency AC. All division I service water pump breakers open which causes isolation of RBCLC and TBCLC. Loss of cooling to the condenser, feedwater, and turbine generator equipment required an immediate shutdown by the operators. Isolation of TBCLC and RBCLC is assumed to result in a low flow trip of the opposite Division service water pumps, thus requiring them to restart. In this sequence, the Division II battery fails (DBI = 6.6E-4) on demand which prevents the restart of Division II service water pumps and prevents the start of Division II safety systems such a RPV injection. All service water is unavailable, feedwater is unavailable, HPCS fails due to loss of room cooling, and RCIC is assumed unavailable due to loss of room cooling. Thus, all injection is unavailable. Procedures for preventing RCIC trip under loss of service water conditions are being developed. Thus, this assumption is conservative and will be considered further in further analyses and updates.

CDF Sequence 4 (1.0E-6/yr)

The initiating event is a loss of Division II emergency AC power (A2X = 4.3E-3/yr) which disables all safety systems that depend on Division II emergency AC. All Division II service water pump breakers open which causes isolation of RBCLC and TBCLC. Loss of cooling to the condenser, feedwater, and turbine generator equipment requires an immediate shutdown by the operators. Isolation of TBCLC and RBCLC is assumed to result in a low flow trip of the opposite Division service water pumps, thus requiring them to restart. The following additional failures occur in the sequence:

- Service water fails (SAH*SBK = 2.9E-4) which causes the loss of condenser and feedwater as well as room cooling.
- Operators successfully open the LPCI "A" and LPCS pump room doors in time to ensure adequate pump cooling and low pressure injection is successful.
- RHR heat removal is unavailable due to loss of service water and containment venting is unavailable due to loss of Division II emergency AC (the nitrogen supply containment isolation valves to the inside purge valves can not be opened without Division II AC power).

Thus, all service water is unavailable and heat removal has failed leading to long term containment failure and then injection failure. No credit is taken for recovering equipment failures associated with divisional AC power and service water over the 20 to 30 hour time frame associated with containment failure. The licensee has indicated that this conservatism will be considered in the future analyses and updates.

The top 100 sequences contributing to core damage frequency are reviewed in Section 3.4. These sequences are reviewed in a number of ways, and a coded (i.e., by abbreviations of system/component failures/successes and operator failures/successes) listing is given for each of the 100 sequences. Section 3.4 also evaluates the key contributors to CDF in a number of ways. including by functional group (ATWS, station blackout, loss of heat removal, loss of injection. and internal floods). Event tree top event importance is reviewed, as are split fraction and human action importance. Common cause and maintenance contributions are indicated for the top 10 sequences.

**In summary, the NMP2 IPE identified the most probable core damage sequences and has identified the dominant contributors to each sequence. Comparisons were made to results developed for other BWRs of reasonably similar designs. These comparisons indicate a higher core damage frequency for NMP2 than was estimated for the NUREG-1150 BWRs. The insights developed regarding dominant contributors for major event categories are generally consistent with those developed for the other BWRs used in the comparison.**

### II.1.2.7 Front-End and Back-End Dependencies

Elements to be considered here include evaluation of sequence screening criteria used, consideration of containment by-pass and containment isolation, plant damage state consideration of reactor system/containment system availability, source term, system mission times, inventory depletion, and dual usage (spray vs injection).

Front-end to back-end interfaces are discussed in several locations in the NMP2 IPE submittal, but Section 3.1.5 specifically addresses this topic. Also, a summary of specific aspects of the Level 1 - Level 2 interface is presented in Section 4.3.3. The methodology used for the NMP2 IPE directly linked the front-end and back-end event trees. Binning of the Level 1 results was performed, but was not needed for the linking. The Level 2 model required information on support system and front-line system availability, RCS conditions (e.g., high pressure, low pressure, LOCA size), reactor power for ATWS events, and containment status. As stated in Section 3.1.5, "Logic rules are specified in terms of top event success and failures and the Level 2 event trees are linked directly to the Level 1 model such that sequences are quantified from initiating event to release category (Level 2 end states). ... The same rules that are used to define Level 1 end states are applied in the Level 2 model. These rules define functional Level 1 sequences based primarily on critical safety function failures. These Level 1 end state rules provide important information about the sequence initiating event type, reactor power, injection systems and containment status. Additional rules are applied to identify whether the RPV is at high or low pressure at the time of core damage as well as whether support and front-line systems and functions are available."

This discussion indicates that important dependency information was transferred directly between the front-end and back-end evaluations for each accident sequence. In addition, the front-end event tree structure clearly indicated that efforts were made to identify and carry along in the sequences information of this type. Several of the front-end tree top events were included specifically for the purpose of tracing front-line and support system status important to the Level 2 analysis.

Based on our review, the following conclusions can be drawn regarding particular aspects of the NMP2 front-end to back-end interfaces and dependency treatment.

Assurance that important sequences were not screened out: The stated cutoff frequency for core damage sequence quantification was 1.0E-10 for all initiating events (Section 3.3.7). All reporting criteria specified in NUREG-1335 relative to screening are stated to have been adhered to. Table 3.4.1-1 of the NMP2 submittal lists core damage sequences with frequencies greater than or equal to 4.0E-8 per year.

Containment bypass considerations: All systemic sequences that contribute to containment bypass frequency in excess of 1.0E-8 per reactor year are identified. The bypass sequences modeled were due to interfacing LOCA events. None of these sequences had a frequency greater than 1.0E-8.

Containment isolation considerations: Containment isolation is the first system nodal event considered in the containment event trees. The conditions or aspects entering into the evaluation/quantification of this node are discussed in Section 4.5.3. Thus, containment isolation is explicitly considered in the NMP2 IPE.

Reactor system/containment system consideration in plant damage states: As noted above, the NMP2 IPE states that the front-end/back-end interface directly linked the Level 1 and Level 2 sequences in a manner which carried along pertinent information regarding reactor system and containment system status for each such sequence.

Source term interface: Both qualitative and quantitative discussions of the source term and associated interface considerations are provided in Section 4.7 of the NMP2 IPE submittal. The MAAP code was the primary deterministic tool used to characterize important accident scenarios. In excess of 50 accident sequence types are stated to have been evaluated using MAAP. Table 4.7-3 presents MAAP calculation results and includes characteristics such as time of fission product release to the containment and fractional release quantities.

System mission times: The sequence transfers from Level 1 to Level 2 are stated to carry along with them considerations of timing, including system mission times.

Inventory depletion: The discussions of the level 1 and Level 2 event tree tops indicates that inventory depletion was considered. Details are not provided.

Dual usage: Dual usage was stated in the submittal to have been considered. Because of the direct linking of the front-end to back-end models on a sequence basis, "accountability of common water sources or common power sources falls out of the combined sequence analysis when it is run from initiating event to release point" (Section 4.3.3).

**In summary, the documentation review indicates that front-end and back-end dependencies were appropriately treated in the NMP2 IPE.**

### II.1.2.8 Consideration of Initiating Events Affecting More Than One Unit; Treatment of Systems Shared Between Units

Nine Mile Point Unit 2 is in close proximity to Nine Mile Point Unit 1 and the James Fitzpatrick nuclear plant. However, it shares no systems with these plants and has separate offsite power supplies.

### II.1.3 Review of the IPE's Quantitative Process

#### II.1.3.1 Quantitative Evaluation of the Impact of Integrated System and Component Failures on Plant Safety; Use of Mean Values and Sensitivity Studies

The NMP2 IPE quantitatively evaluated the impact of integrated system and component failures on plant safety. The quantification process used to estimate unavailabilities of systems and functions is described in Section 3.3.5. That section also lists the mean value of the event tree top split fractions used in the sequence quantification. The process used for quantification of sequence frequencies is discussed in Section 3.3.7.

The IPE submittal states that fault trees were developed to model systems represented in the event tree top events and to develop split fraction values. The system descriptions provided in Section 3.2 state that the fault trees are included in the Tier 2 documentation, and are available on request. No listing of the fault trees used in the NMP2 IPE evaluations was provided in the documentation. However, all split fractions for the systems analyzed are presented in Table 3.3.5-1.

The discussion in Section 2.1 of the overall methodology used to perform the NMP2 IPE implies that component failure frequencies with individual confidence bounds were used, and that Monte Carlo simulation was used to account for the various individual sources of uncertainty to determine the confidence bounds on the overall results. However, the discussion does not explicitly state that these simulations were actually performed for the NMP2 IPE.

The accident sequence quantification process utilized mean values for the initiating event frequencies and event tree top event unavailabilities (Section 3.4.1).

Level 1 sensitivity analyses were presented in Section 3.4.2 in the form of Importance Analyses. Importance is a characterization of the significance of an analysis element to core damage frequency. Other importance measures (risk reduction ratio and risk achievement worth) are measures of the sensitivity of the results to changes in particular parameters (e.g., failure rates, system unavailabilities, etc.). The NMP2 IPE assessed importance of initiating events, event tree top events, split fractions, and human actions. The 50 most important event tree top events are ranked and presented in Tables 3.4.2-1 and 3.4.2-2. The results of this importance assessment suggested the following ranking of system importance:

- Emergency AC Power
- AC Power Recovery
- RHR System
- Containment Venting
- High Pressure Core Spray
- Service Water System
- Containment Failure Causes Core Damage
- Reactor Core Isolation Cooling
- DC Power
- Emergency Core Cooling System Pump Room Cooling

The evaluations performed calculated the impact on CDF when the parameter or top event of interest was set to "guaranteed failure or success".

Other than the importance analyses noted here, the NMP2 IPE did not document or cite the results of any Level 1 evaluations to determine the impact of vital assumptions.

**In summary, the NMP2 IPE quantitatively evaluated the impact of integrated system and component failures on plant safety. Mean values were used to quantify accident sequences. Sensitivity studies were presented in the form of Importance Analyses.**

### II.1.3.2 Consistency of Techniques Used to Perform Data Analysis

The technique used to perform data analysis was the PLG approach to Bayesian analysis (IPE Ref. 37). One-stage or two-stage Bayesian analysis was used to combine data from up to three sources: expert opinion generic data, industry-wide generic data, and plant-specific data. The submittal contains details of the procedure. The technique is consistent with other PSAs.

**Plant-Specific Data**

Although the plant began commercial operations relatively recently (April 1988), the licensee had collected plant-specific data for most component types, reported these data in the submittal, and used them for Bayesian update of the generic data. Sources were the Nine Mile Point Unit 2 Inservice Testing database and the Nine Mile Point Unit 2 Nuclear Plant Reliability Data System. Components were grouped by type rather than system (centrifugal pump, reciprocating pump, etc.), although the turbine-driven pump data is specific to the RCIC system.

Plant-specific data were used for most of the important components and systems as identified in NUREG-1335. The exceptions were demand failures of batteries or circuit breakers because the number of demands for each of these components was not available.

For maintenance unavailability, the INPO Quarterly Performance Indicator Data report for Nine Mile Point Unit 2 was used.

**It is concluded that the licensee has met the NUREG-1335 reporting guidelines for data analysis techniques. The techniques used are consistent with those used in other PSAs and plant-specific data are used, to the extent available, for important components and systems and for maintenance unavailabilities.**

### II.1.3.3 Sources of Generic Failure Data and Rationale for Their Use

The licensee identified the sources of generic data for transient initiating event frequencies to be LER data obtained for all BWR plants except NMP2 that were commercially operating during the period of 1984 through 1990. The licensee decided to use the more recent operating experience because both this data and INPO evaluations showed significant reductions in number of events per year as compared to earlier years, which were attributed in part to industry efforts to reduce scrams and improve overall plant performance.

For LOCA inside containment, the event frequencies were derived from the BWR EPRI evaluation that was performed as part of the Shoreham PRA (IPE Ref. 15). The submittal compares various estimates of large pipe rupture failure rates from various sources and notes that there is an appreciable overlap in the error bounds. A survey of seven other BWR PRAs shows that the EPRI frequencies for large and intermediate LOCAs are at the upper end of the values previously used. The frequency for small LOCAs is within the range of values previously used, if reactor recirculation pump (RRP) seal LOCAs are omitted.

WASH-1400 and the EPRI ISLOCA evaluation (IPE Ref. 45) were used for generic data relevant to LOCAs outside containment.

This review examines other usage of generic failure data as parts of the analyses of Sections 3.3.1 and 3.3.8 of the NMP2 IPE submittal. Because support system transient initiators are calculated from support system fault trees, their "generic data" are those that are used to develop component failure probabilities.

The licensee identified the primary sources of generic data to be the PLG proprietary database (IPE Ref. 39) and NUCLARR (IPE Ref. 40). The PLG database, based on and evolved from PRAs performed by PLG and on data collected from U.S. reliability data sources, provided the basis for expert opinion generic data. NUCLARR, being a compilation of actual component failure records in the nuclear industry, provided the basis for industry-wide generic data. For most component failure modes, data were available from both sources and were combined in a one-stage Bayesian analysis to produce a generic database for the IPE in the form of probability distributions that characterize the uncertainty of the data. Other sources for generic component data were IEEE STD-500 and NSAC-152 (Peach Bottom Unit 2 PRA), which were used in the few instances that no data were available from the primary sources.

The PLG database has already been reviewed by the NRC (NUREG/CR-5606). The reviewers found that the database was extensive. A comparison of the PLG database with the data contained in NUREG-2815, Appendix C indicates reasonable consistency.

The licensee identified the PLG database as the source of generic maintenance unavailabilities.

The licensee identified Nuclear Power Experience (IPE Ref. 39) as the source of generic data for internal flood initiating frequencies. This reference contains industry data concerning actual occurrences of flooding at nuclear power facilities.

**Considering also the information provided in Sections 3.1.1 and 3.3.1 of the submittal, we conclude that the licensee has explicitly identified the sources of generic data and has provided the rationale for the choices. One of the primary sources has been reviewed and accepted by the NRC, and it is reasonably consistent with the data contained in NUREG-2815, Appendix C.**

### II.1.3.4 Common Cause Failure Data and Data Sources

The licensee identified the PLG generic common cause database (IPE Ref. 39) as the source of generic component data for common cause parameters. The generic data are in terms of probability distributions which characterize the uncertainty in the parameters. Plant-specific data were not used because of limited plant experience and the relatively short time that the plant has been in operation.

**We conclude that the licensee has addressed the common cause failures in the analysis, including the process and sources of data used per the NRC IPE guidance.**

### II.1.4 IPE Approach to Reducing the Probability of Core Damage or Fission Product Release

### II.1.4.1 Core Damage Vulnerability and Efforts to Uncover Vulnerabilities; Plant Modifications (or Safety Enhancements) to Eliminate or Reduce the Affect of Vulnerabilities

The NMP2 IPE submittal did not provide a specific definition of plant vulnerability. However, the discussion in Section 3.4.2 entitled "Vulnerability Screening" presents the discussion of vulnerabilities in terms of core damage frequency. The NMP2 IPE defined core damage events as those instances wherein two thirds of the active core length would remain uncovered for more than a brief period of time, i.e., the reactor vessel water level subsequently not restored. Based on the discussion provided in the IPE submittal and on subsequent responses from the licensee to NRC questions, the NMP2 criteria used in the screeing process for Level 1 vulnerabilities was a calculated CDF of 1.0E-4 or greater per year.

Section 3.4.2 states that their evaluations uncovered no unusual or plant unique contributors to core damage compared to what has been found in PSAs for other BWR plants.

The NMP2 IPE provided an examination of core damage frequency results in a number of ways to develop plant-specific insights of the importance of systems, functions, and human actions. These screening evaluations, presented in Section 3.4.2 of the submittal, discussed "importance" of contributions of functionally grouped sequences (e.g., ATWS, station blackout, loss of heat removal, loss of injection, and internal floods), of initiating events, of event tree top events. of split fractions, of contributors to split fractions, and of human actions.

For each of these "importance" groups or classes, the submittal reviewed the key contributors in terms of functional or system failures that resulted in failure to provide adequate core cooling. The discussions provided indicate that the licensee put forth an effort to understand what the key contributors to failure were, and to understand what systems and operator actions were most important to reducing plant vulnerabilities. To accomplish this, the IPE submittal ranks the importance of the top 50 event tree top events (Table 3.4.2-1 and 3.4.2-2), it ranks the split fractions by importance and risk reduction worth (Table 3.4.2-3) and by risk achievement worth (Table 3.4.2-4). Similarly, the split fractions with human actions are ranked for importance in Table 3.4.2-6. Contributors to important split fractions, including important support systems, are listed in Table 3.4.2-7.

These vulnerability or importance assessments were carried out to the event tree top event level. This level typically represented success or failure of system trains or segments; thus, the NMP2 IPE went beyond the system level in attempting to assess vulnerabilities.

That the IPE was carried out to the train and component level is also indicated by the detail provided in the system descriptions (Section 3.2). Each system description typically included a Component Block Description which tabulated component identification numbers and descriptions, their failure modes, initial and actuated states (open, closed, standby, etc.), identification of support systems, and a description of what occurs upon loss of support.

The approach taken in the NMP2 IPE to identify plant vulnerabilities appears to be reasonable, and it provided a means for identifying particular systems or particular operator actions which, if improved, could reduce the potential for core damage events at the plant.

**In summary, it is our judgment that the analysis supports the licensee's assessment of vulnerability with respect to core damage. The analysis technique included consideration of potential failures beyond the train level, with consideration given to system dependencies and interrelationships between and among the initiating events and various plant systems.**

### II.1.4.2 Identification of Plant Improvements and Proposed Modifications Expected to Enhance Plant Safety

The evaluation of insights for plant improvements developed from the IPE process is described in Section 6. That section reviews noteworthy NMP2 safety features, it discusses planned improvements to the plant configuration and operating procedures that resulted from IPE-developed insights, and it discusses other insights which may be implemented at a later time, depending on the outcome of further evaluations. Among the planned improvements to the plant are:

Modifications to the containment venting system (installation of valves in the standby gas treatment system). This modification is to be installed during the 1993 refueling outage. In addition; operating procedures related to containment venting are being developed. These modifications will provide higher confidence for successful containment venting when required.

Development of procedures to enhance Auxiliary Bay pump room cooling during loss of service water accidents. The implementation of these procedures should reduce the likelihood of loss of HPCS, RCIC, and low pressure injection pumps due to loss of room cooling due to loss of service water.

Enhancement of station blackout procedures. The IPE took credit for the use of improved procedures for dealing with station blackout scenarios. The changes include procedural changes to bypass RCIC isolation interlocks, shedding all non-essential DC loads within the first 2 hours of the event, procedures to promote use of fire water pump injection through the RHR, guidance for operation of SRVs to minimize depletion of nitrogen and DC power, and guidance for local closure of outside containment isolation valves.

Enhancement of procedures for dealing with internal flood scenarios. Procedures are to be developed to help operators diagnose and mitigate internal floods.

Improved test and maintenance procedures to reduce the likelihood of an interfacing system LOCA.

The foregoing improvements were identified by the IPE and are consistent with the vulnerability screening and importance evaluations discussed in Section 3.4 of the submittal. The documentation indicates that the licensee is in the process of implementing these changes.

The NMP2 IPE submittal states that the analysis model used took credit for some, but not all, of the foregoing improvements. However, the analysis results did not present a before and after picture, i.e., no quantitative measure of the risk reduction achieved by implementing these changes was presented. Similarly, there is no evaluation of the impact on overall risk of implementation of the remaining improvements. The NMP2 IPE submittal would have been enhanced if the licensee would have provided this type of quantitative evaluation of risk reduction due to changed in plant configuration and operation.

**Based on our review, we conclude that the licensee has taken reasonable action in response to the results of their assessment. More specifically, the licensee has identified both physical and procedure modifications that are expected to enhance plant safety. The IPE could be improved through the quantitative evaluation of risk reduction resulting from the implementation of the specified improvements in plant configuration and operation.**

### II.1.5 Licensee's Evaluation of the Decay Heat Removal Function

#### II.1.5.1 Reliability of the DHR Function and Consistency With Other PSA Findings

Section 3.4.3 of the NMP2 IPE presents the decay heat removal assessment. As stated in this section, the systems or functions that can provide successful decay heat removal in the IPE model are as follows:

- Main Condenser
- RHR System
- Containment Venting
- Continued Injection following Containment Failure

The event tree top events that model these important means for accomplishing decay heat removal are discussed. The discussion noted that loss of injection can result in loss of decay heat removal; however, loss of injection is stated to be a more immediate concern and is more likely to cause core damage much earlier in time relative to loss of long term decay heat removal. The licensee stated that the assumption was made that loss of decay heat removal is concerned with the long term loss of DHR. Therefore, their evaluation presented in Section 3.4.3 did not include the loss of injection sequences in the decay heat removal evaluation.

Loss of decay heat removal sequences are stated to account for about 29% of the total CDF.

The importance of event tree top events to total decay heat removal core damage frequency is provided in Table 3.4.2-7. The licensee's assessment from this evaluation is that containment venting and RHR pump trains A and B have the greatest importance to achieving successful decay heat removal, and states that any improved availability of these systems would provide the greatest reduction in the loss of decay heat removal sequences. Further, service water trains A and B, and emergency AC Division II are stated to be the most important support systems.

Split fraction importance for the top events noted above are listed in Table 3.4.3-2. This table helps to identify the support systems and functions having the greatest impact on accomplishing the decay heat removal function. The documentation states that improving the reliability of these systems and equipment will reduce the frequency of loss of decay heat removal. Beyond that, however, there is no elaboration as to how reliability improvements might be achieved.

The NMP2 features for accomplishing decay heat removal are similar to those of other BWRs, including the NUREG-1150 plants Peach Bottom and Grand Gulf.

### II.1.5.2 IPE Consideration of Diverse Means of Decay Heat Removal

The NMP2 IPE submittal reviewed the alternative means of accomplishing decay heat removal at Nine Mile Point Unit 2. These were discussed in Section II.1.5.1 above. In addition, the submittal discussed alternative DHR means such as fire water and service water system crossties to the RHR system.

### II.1.5.3 Decay Heat Removal Unique Features

There was no discussion of plant specific unique features relative to accomplishing the decay heat removal function. Noteworthy NMP2 safety features are identified in Section 6.1, and several of these features enhance the reliability of removing decay heat. Among the features cited were:

- Spatial separation of auxiliary bays and the use of submarine type doors to the auxiliary bay pump rooms, HPCS, and RCIC provide substantial protection from floods and other hazards. These features enhance the DHR function reliability for certain scenarios and sequences.
- The HPCS is independent including actuation system inputs and emergency AC. However, this system is unavailable during station blackout events because the service water system is unavailable, and is needed to cool the HPCS diesel. A potential improvement to reduce this vulnerability has been identified.
- NMP2 has a hardened containment venting capability.

### II.1.5.4 Conclusion Regarding NMP2 Evaluation Decay Heat Removal

**The NMP2 IPE discussions of the DHR function, its reliability, and alternative means of accomplishing DHR show that the IPE is capable of identifying vulnerabilities associated with decay heat removal. While these assessments generally identified the support system or front-line systems with the greatest impact on the DHR function, they did not address specific changes to the plant design or operating procedures which might enhance DHR reliability. A more specific discussion of such improvements would be helpful. No vulnerabilities were identified.**

## III. OVERALL EVALUATION AND CONCLUSION

Based on a review of the NMP2 IPE submittal, it is our judgment that this submittal is consistent with the requirements of NUREG-1335 and is consistent with the methods identified in Generic Letter 88-20. The NMP2 IPE provides assistance in understanding the actions needed and options available for prevention and mitigation of severe accidents, and for identifying plant vulnerabilities.

The NMP2 IPE employed the large event tree/small fault tree approach. This IPE used a fully integrated coupling of the front-end and back-end analysis. Steps were taken to assure that the evaluation was performed on the as-built, as-operated plant as it existed at the completion of the first operating cycle (specific date for this cycle was not provided). However, the models employed in the NMP2 IPE took credit for some physical modifications and procedural changes which have not yet been implemented. Since the impact of these changes on CDF was not quantitatively evaluated, their importance to the IPE conclusions cannot be assessed.

The internal flooding analysis performed is judged to be sufficient to identify the dominant flooding core damage sequences.

The IPE treatment of initiating events is generally satisfactory. However, there are some weaknesses as the licensee did not provide the rationale for its consolidation of given initiating events into particular categories of initiating events.

The review of the event trees developed and used in the NMP2 IPE indicated that they were sound and well thought out. Additional explanation of the rationale and specific criteria used in developing the support system event tree, including the treatment of system asymmetries, would enhance the IPE documentation..

The submittal included an importance analysis. The impact of sensitive assumptions was not explicitly treated other than through the importance analyses.

It is concluded that the licensee has met the NRC's review guidance criteria for data analysis techniques. The techniques used are consistent with those used in other PSAs and plant-specific data are used, to the extent available, for important components and systems and for maintenance unavailabilities.

The licensee did not provide an explicit definition of vulnerabilities. Such a definition should be provided to remove possible ambiguities. Based on our review, we conclude that the licensee has taken reasonable action in response to their assessment. More specifically, the licensee has identified both physical and procedure modifications that are expected to enhance plant safety. The IPE could be improved through the quantitative evaluation of risk reduction resulting from the implementation of the specified improvements in plant configuration and operation.

The NMP2 IPE discussions of the DHR function, its reliability, and alternative means of accomplishing DHR show that the IPE is capable of identifying vulnerabilities associated with decay heat removal. Contributors were identified in the form of importance assessments. While these assessments generally identified the support system or front-line systems with the greatest impact on the DHR function, they did not address specific changes to the plant design or operating procedures to enhance DHR reliability. A more specific discussion of such improvements would be helpful.

ENCLOSURE 3

NINE MILE POINT NUCLEAR STATION, UNIT 2 INDIVIDUAL PLANT EXAMINATION

TECHNICAL EVALUATION REPORT

(BACK-END)