CA/TR-92-019-07

TECHNICAL EVALUATION REPORT

NINE MILE POINT UNIT 2
INDIVIDUAL PLANT EXAMINATION

ASSESSMENT OF HUMAN RELIABILITY ANALYSIS
STEP 1 REVIEW

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1. INTRODUCTION

This technical evaluation report (TER) is a summary of the Step 1, documentation-only review of the Human Reliability Analysis portion of the Nine Mile Point Unit 2 Individual Plant Examination (IPE) submittal to the U. S. Nuclear Regulatory Commission (NRC). The body of the report consists of four sections, per the instructions of the Task Order: (1) this Introduction, which provides a brief summary of the approach to this Step 1 review and of the Nine Mile Point Unit 2 IPE HRA approach; (2) Contractor Review Findings, a detailed documentation of findings for each work requirement specified in the Task Order; (3) overall Evaluation and Conclusions, which summarizes the important findings and results from the review, and (4) the NRC summary data sheets.

1.1 Step 1 HRA Review Approach

The Step 1 review approach for Nine Mile Point IPE HRA involves the following six steps illustrated in Figure 1. These steps, especially steps 2 through 4, are interactive and iterative, but follow this general progression:

(1) **Scoping Review** - an overview of the entire IPE submittal. Read summary sections, plant descriptions, the major HRA-pertinent section(s), and result sections; skim/scan the entire submittal, including appendices and detailed front-end and back-end analyses. Identify the basic approach used for the HRA; the organization of the HRA documentation, including any obvious major omissions; notable features of the plant, the overall IPE approach, or the HRA approach that deserve special attention; references that may need to be reviewed or checked; obvious points of interface with front-end and back-end analysis. Review descriptions of IPE/HRA team qualifications.

(2) **Detailed Review of HRA Sections** - a detailed review and assessment of the primary HRA section(s) of the submittal. This involves first a thorough (re)reading of descriptions of methodology, noting assumptions, data sources, and other important aspects of the analysis, and annotating any questions, potential problem areas, missing information, or issues for further investigation. Second, it involves a comparison of information and documentation found in the submittal about the overall HRA methodology/approach to the information/ documentation "requirements" identified in accepted HRA approaches used in other PSAs. For example, since one of the HRA methods used in the NMP2 IPE was the Technique for Human Error Rate Prediction (THERP), this comparison involved reviewing the information contained in the submittal regarding the major steps in the THERP approach as described in NUREG/CR-1278 (Ref. 1) for HEPs evaluated with THERP. Finally, the detailed review involves an attempt to "track" the complete assessment of a few key operator actions through the HRA process described in the submittal. By tracking, we mean simply identifying that the submittal contains sufficient information to clearly delineate methodology, major assumptions, important parameters such as performance shaping factors, data sources. and references for both the qualitative and quantitative assessment of human actions. There is no attempt to reproduce quantitative analysis.

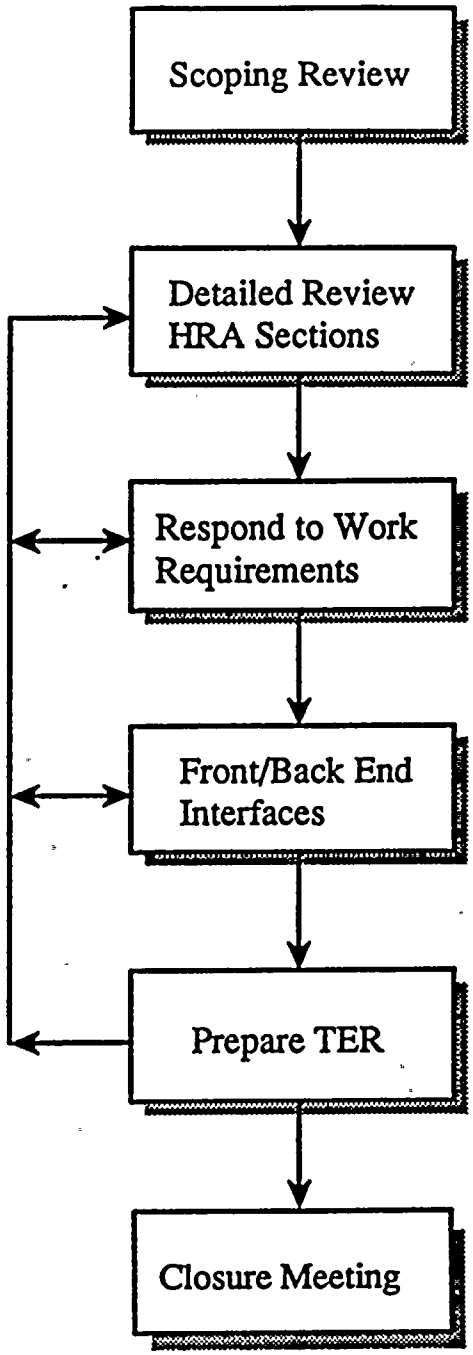

Figure 1 - Human Reliability Analysis Step 1 Review Approach

(3) **Response to Work Requirements** - assessment of specific issues identified in the Task Order work requirements. This is an item-by-item assessment responding to each work requirement. The focus is identification of strengths and weaknesses of the HRA portions of the submittal and insights regarding important results or potential areas of improvement. Any questions that require additional input from the licensee are identified. This step includes completion of the NRC Data Sheets, which is Work Requirement 2 in the Task Order.

(4) **Interface with Front-End and Back-End Reviewers** - two-way exchange of information and discussion of issues. The focus is on HRA aspects of front-end or back-end analysis, but includes a general exchange of information and findings. The interaction takes place informally throughout the review, but primarily after completion of the overview in Step 1 above, and again after completion of Steps 2 and 3 as writing of the TER begins. More formal interaction occurs during the closing meeting of NRC staff and IPE review contractors in Step 6.

(5) **Prepare the TER** - develop and write this technical evaluation report. This involves: preparation of a draft report documenting all work accomplished, findings, and conclusions; internal technical review verifying findings and conclusions, and compliance with Task Order requirements; and, editorial review and printing.

(6) **NRC Staff and Contractor Meeting** - held after submittal of the TERs from contractors to review findings and conclusions and finalize questions for the licensee (if any).

1.2 Nine Mile Point IPE HRA Approach

The Nine Mile Point Unit 2 (NMP2) IPE consists of a Level 2 Probabilistic Risk Assessment (PRA) which follows the overall methodology described in the PRA procedures guide, NUREG/CR-2300 (Ref. 2) using large event trees linked from initiating event to containment response and radionuclide release for sequences which lead to core damage. Event Sequence Diagrams summarizing system and operator actions necessary for successful response to initiating events were constructed along with subsequent event trees. The event trees are based on evaluation of plant systems, intersystem dependencies, equipment response to plant events, operator response to plant events, and success criteria. Specific operator actions from the emergency operating procedures (EOPs) and supporting procedures are incorporated as operator response to plant events. MAAP analysis was used to provide transient behavior of important plant parameters and timing and success criteria used in event tree evaluation, including operator actions.

Human Interaction (HI) events analyzed were identified from the event trees. Functional grouping of scenarios (in terms of cues, procedures, and key operator responses) for human interactions was performed to reduce the number of evaluations. A human error probability (HEP) was evaluated for the most demanding scenario in a functionally similar group. The limiting or bounding scenarios were identified through discussions with the event sequence analysts. The front-end and back-end HRA analyses were performed by different analysts, and different levels of detail were provided in the submittal for the two analyses. The HRA

approach described in the submittal, essentially directed at quantifying human error probability (HEP) estimates to be incorporated in the Event Trees, was performed using three methodologies: (1) EPRI (Ref. 3); (2) THERP (Ref. 1); and (3) ASEP (Ref. 4). Some plant-specific data was obtained from the simulator and operator and trainer interviews, but no details were provided for other data sources in the submittal.

2. CONTRACTOR REVIEW FINDINGS

The subsections below address explicitly, item by item, each of the work requirements specified in the Task Order. For each item, there is an attempt to identify notable points about the submittal, both strengths and weaknesses, and to provide insights as to how the submittal might be improved with regard to the specific work requirement and to the overall intent of Generic Letter 88-20. If there are requests for additional information for the licensee, those questions are identified in bold text.

2.1 Work Requirement 1.1 Perform a general review of the human reliability analysis.

2.1.1 WR 1.1.1 The IPE submittal is essentially complete with respect to the type of information and level of detail requested in the IPE Submittal Guidance Document NUREG-13435. List any obvious omissions.

Table 2-1 lists the major items identified in NUREG-1335 pertinent to HRA that were checked. The following are the findings for this work item:

(1) General Methodology. The overall description of the HRA effort provides a reasonably clear understanding of the general methodology and approach to addressing human actions within the IPE. Different analysts performed the front-end and back-end HRA analysis. The submittal states that a concerted effort was made to ensure consistency between the two HRA analysis. The back-end analysis provided more detailed information about the process followed and the individual HI evaluations. The model of human interactions used for the evaluation of HEPs is a simple one that splits the response into two phases: detection, diagnosis and decision (DDD); and execution. Three methods were used to evaluate the operator actions in the IPE: (1) EPRI; (2) THERP; and (3) ASEP. The application and methodology are discussed in sections 3.3.3 and 4.6.2.5 of the submittal for front-end and back-end HRA analysis respectively. No detailed description of the application of the three methods was provided.

Both pre-event and post-event human errors were evaluated for the IPE. Data sources referenced for example calculations were simply the basic references for THERP and ASEP. No specific information pinpointing precise tables, etc. was provided. Other data was obtained from NMP2 simulator observations, interviews with trainers and operators, and from industry and INPO LER data, but no details were provided in the submittal for the HRA analysis.

Event Sequence Diagrams summarizing system and operator functions necessary to successfully respond to initiating events were constructed. Event trees were produced from the Event Sequence Diagrams taking into account potential failures. Specific operator actions from the EOPs and other procedures are incorporated along with automatic system actions in the event trees. The event trees contained the function which the operator actions were to accomplish. The HRA analysts used procedures, operator and trainer interviews, and simulator observations to complete the information required for the HRA analysis of the operator functions in the event trees. Functional grouping of scenarios (in terms of cues, procedures, and key operator responses) for HIs was performed to reduce the number of evaluations. An HEP was evaluated

Table 2-1 NUREG-1335 HRA Items Checked - WR 1.1.1

NUREG-1335 REFERENCE	INFORMATION PERTINENT TO HRA
2.1.1 General Methodology	Concise description of HRA effort and how it is integrated with the IPE tasks/analysis.
2.1.2 Information Assembly	2.1.2.2 List of reference PRAs, insights regarding HRA, human performance. 2.1.2.3 Concise description of plant documentation used for HRA information; concise discussion of the process used to confirm that the HRA represents conditions in the as-built, as-operated plant. 2.1.2.4 Description of the walkthrough activity, including HRA specialist participation.
2.1.3 Accident Sequence Delineation	Description of process for assuring human actions considered in initiating events and accident sequence delineation; HRA specialist involvement.
2.1.4 System Analysis	Description of process for assuring that the impacts of human actions are included in systems analysis; process for integrating HRA.
2.1.5 Quantification Process	2.1.5.1 HRA in common cause analysis. 2.1.5.3 Types of human failures considered in the IPE; a categorization and concise description exist. 2.1.5.4 List of human reliability data and time available for recovery actions; data sources clearly identified; if screened, a list of errors considered, criteria for screening, and results of screening. 2.1.5.5 List of HRA data obtained from plant experience and method/process for obtaining data; list of generic data. 2.1.5.6 Concise description of method by which HEPs are quantified, including break down such as task analysis, and techniques for combining probabilities, assessing dependencies, etc.

Table 2-1 NUREG-1335 HRA Items Checked - WR 1.1.1

NUREG-1335 REFERENCE	INFORMATION PERTINENT TO HRA
2.1.6 Front-End Results and Screening Process	Human contributions to important sequences are clearly identified. A concise definition of vulnerabilities is provided, along with a discussion of criteria used to identify vulnerabilities. A listing of vulnerabilities is provided, with clear definition of those related to human performance. Underlying causes of human related vulnerabilities are identified. 2.1.6 Sequences that, were it not for low human error rates in recovery actions, would have been above the applicable core damage frequency screening criteria are identified and discussed. 2.1.7 Any human performance issues pertinent to USIs or GSIs are identified and discussed as appropriate.
2.2 Back-End Submittal	Impacts of operator action on containment response are identified. Actions assumed to be accomplished by operators can reasonably expected to be accomplished under the severe accident conditions expected; equipment accessibility, survivability, information availability, etc. have been considered. Critical human actions have been identified and included in the event trees and quantitative HRA assessments.
2.3 Specific Safety Features and Potential Improvements	Any human performance related aspects of unique and/or important safety features are discussed, including any that resulted in significantly lowering typically high frequency core melt sequences. Human related potential improvements - procedures, training, etc.- in response to vulnerabilities are clearly identified and discussed.
2.4 IPE Utility Team and Internal Review	The submittal describes the utility staff participation and involvement in the HRA. An independent in-house review of the HRA was conducted.

for the most demanding scenario in a functionally similar group. The limiting or bounding scenarios were identified through discussions with the event sequence analysts.

(2) Information Assembly. The submittal described a listing of reference PRAs of similar plants that were reviewed by the licensee for the NMP2 PRA. Discussions on the results of each of the reviewed plants was provided. Plant documentation used to acquire HRA information was listed and described, including: plant operating procedures, EOPs, and surveillance and maintenance procedures. In addition, interviews with operators and trainers, and simulator observations were sources of information for the HRA evaluation and were included as a part of the walkdown process. The NMP2 support team included operations and training personnel as well as EOP developers to provide information on the operation of the plant during normal and accident situation, and to discuss operator actions and training. HRA analysts also participated in system walkdowns. There is no information provided regarding the structure used to collect or assess information from walkdowns and operator and trainer discussions.

It is recognized that there is a great deal of information to be assimilated, and much of it may be too detailed for incorporation into the IPE submittal. It is also recognized that some of the information for the HRA is derived from interviews and discussions rather than formal documentation. However, the IPE submittal would be strengthened if specific information were provided to demonstrate that the process for obtaining information from operators and trainers and from plant walkdowns was systematic, structured, and rigorous.

(3) Accident Sequence Delineation. Identification and delineation of initiating events is discussed in Section 3.1 of the submittal. A detailed discussion of the process used and information sources for identifying potential initiating events are included. For each event, a discussion of the initiators, automatic system responses, major assumptions, and operator functions are included. The description of the initiating events is comprehensive and appropriately detailed for the submittal. With regard to HRA concerns, it appears that the process approximately considered human error contributions to accident initiation and sequence progression.

(4) System Analysis. The System analysis is described in Section 3.2 of the submittal. System descriptions are appropriately detailed. In addition to routine information on major components and instrumentation, they include system dependencies and interfaces, testing and maintenance, technical specifications, system operation, modeling assumptions, and success criteria for each front line system. Human actions are incorporated in front-line event trees as top event functions and in system fault trees when only the system is affected. Documentation of system fault trees is a part of the Tier 2 documentation and should provide sufficient information for a detailed evaluation if necessary.

(5) Quantification Process. The submittal contains a listing of results (calculated HEPs) from the quantification process, but very little details on methodology or "input" data. Three methods were used to evaluate the operator actions in the IPE: (1) EPRI methodology from Ref. 1; (2) THERP; and (3) ASEP. The application and methodology are discussed in sections 3.3.3 and 4.6.2.5 of the submittal for front-end and back-end analysis, respectively. The back-end analysis description included the method used for evaluating each of the operator actions, but no details were provided on use of PSFs, data sources, etc. Examples provided in the front-end analysis were reviewed for application of the ASEP and THERP methods.

Pre-initiator, e.g., maintenance errors in testing, equipment calibration, and restoration were addressed in the IPE. Industry and plant specific data (including procedures) and qualitative screening criteria were used to conclude that only SLS and ECCS required evaluation. Details of the pre-initiator event discussion in the submittal is provided in Section 2.1.2. The qualitative screening criteria appear to be appropriate.

(6) Front-End Results and Screening Process . Front-end results, including internal flooding, and the screening process are reported in Section 3.4 of the submittal. Vulnerabilities are discussed in Section 2.4.1 of this TER. The screening criteria used for reporting event frequencies and core damage frequency were taken from NUREG-1335 and GL 88-20.

There were a number of enhancements involving human reliability which were identified and included in the results of this IPE. In addition, "IPE insights" were identified which are under consideration for implementation. These are discussed in greater detail later in Section 2.4.2 of this TER.

A strength of the IPE is the incorporation of the EOP logic and actions into the event sequences, including containment event trees. EOP operator actions were evaluated for severe accidents, including post core damage events, to determine their effectiveness. Insights were identified and discussed in Section 6.3 of the submittal which will be useful for development of Accident Management Program or improvement of EOPs with respect to severe accidents.

Split fractions which include human actions are addressed in Section 3.4.2.5 of the submittal. The risk reduction worth for top event split fractions with human actions was provided. Many of the split fractions also contained equipment failures. The contribution of human actions cannot be ascertained from information provided. No discussion of the input of human interaction on sequences of events or CDF was provided.

(7) Back-End Submittal. The Level 2 analysis includes important human actions that can affect containment performance and radionuclide release frequency, magnitude and timing. In general the actions considered were from the EOP actions but some non-EOP recovery actions were included. Notable as a strength of the NMP2 IPE is the incorporation of the latest approved EOP guidance in the Containment Event Trees to evaluate the containment response. This provides an evaluation of the EOP guidance for operators during severe accidents where the containment function is threatened. The Level 2 analysis was evaluated and Accident Management insights were developed in Section 4.8.12 of the submittal.

(8) Specific Safety Features and Potential Improvements. A number of specific safety features of NMP2 were discussed in Section 6. Specific procedure changes are being implemented and are included in the results of the submittal. These include:

- Containment vent modifications and procedure improvements to manually open outside valve and to align instrument air when nitrogen is not available.

- Opening doors for pump room cooling with loss of room cooling.

- Station blackout procedure to provide: bypassing RCIC isolation circuitry within 2 hours; load shedding nonessential DC loads within 2 hours; remote lineup of diesel fire pump to RHR for injection flow; instructions for depressurization to conserve nitrogen and D C power and enhance use of RCIC or diesel fire pump; and local closure of containment isolation valves with loss of AC power.

- Procedural guidance to open outside doors for service water or fire water flooding to divert water out of building.

- Add precautions to test and maintenance procedures to ensure no inadvertent opening of low pressure injection paths during power operation reducing likelihood of interfacing system LOCA.

Improvements or enhancements under consideration include:

- Standby Liquid Control System (SLS) is made inoperable during testing. Test procedures were identified to be revised for recovery of inoperable system should SLS initiation occur during testing. An additional enhancement is to add to the chemistry surveillance procedures a requirement for independent verification of air isolation valves closed.

- Section 6.3 in submittal provides a group of EOP technical issues which will affect execution of EOPs.

- Table 6.1-1 in submittal provides details of recommended plant improvements.

- Accident Management Insights were developed in Section 4.8.12 of the submittal.

No discussion of the improvements in the results for CDF was provided.

(9) IPE Utility Team and Internal Review. A group of 5 full time engineers was assembled in 1990 as a permanent PRA organization. The IPE team included an individual responsible for HRA analysis. It appears that most of the HRA analysis was done by consultants, but this was not specified in the submittal. The support team assembled for the HRA analysis included many personnel from Operations familiar with EOP development and implementation as well as training personnel. Thus, HRA analysts had available to them individuals who had plant-specific information on EOP execution and operator training. These individuals also provided a means for feedback of HRA analysis to plant staff. The formation and support of this permanent organization represents significant management commitment to the PRA/IPE effort.

An independent review team was headed by the QA organization and the Independent Safety Engineering Group (ISEG). IPE review team members and organizational responsibilities were provided in Tables 5-5 and 5-6 of the submittal. Responsibility for the HRA review was divided between QA, ISEG and Operations reviewers. The qualifications of the HRA reviewers was not detailed in the submittal.

2.1.2 WR 1.1.2 The employed HRA methodology is clearly described and justified for selection.

The submittal contains a concise listing of results (calculated HEPs) from the quantification process, but very little details on application of the methodologies or "input" data. Times available for operator action appear to have been analyzed using MAAP and operator feedback. The model of human interactions used for the evaluation of HEPs is a simple one that splits the response into two phases: detection, diagnosis and decision (DDD); and, execution. Three methods were used to evaluate the operator actions in the IPE: (1) EPRI methodology (Ref. 3); (2) THERP (Ref. 1); and (3) ASEP(Ref 4). The application and methodology are discussed in sections 3.3.3 and 4.6.2.5 of the submittal for front-end and back-end analysis respectively. A description of when each type of methodology is used is provided. By working through the examples provided, the application of ASEP and THERP could be followed. It should be noted that the reference for the example in Section 3.3.3.2.1 was incorrectly cited as the EPRI model, but it was obviously the ASEP reference. **The submittal would be strengthened by a more detailed description of the application of THERP and ASEP methods, including sources of human error probability data, any plant-specific PSFs and the rationale for selection of PSFs for each type of human action evaluated.** None of the examples provided any information on the application of the EPRI methodology.

The EPRI methodology document, "An Approach to the Analysis of Operator Actions in Probabilistic Risk Assessment," EPRI-TR-100259, was not available for this assessment. A copy of this reference document and a discussion of the Licensee's use of the reference is necessary to assess the reasonableness of the approach, data, and assumptions to arrive at the listed HEPs.

The back-end analysis description included the method used for evaluating each of the operator actions in the table of results, but no details were provided on use of PSFs, data source, etc. The discussion included a description and sequence of the tasks to perform the HRA analysis. The statement is made in the submittal that the efforts were coordinated to ensure that the HRA analysis was consistent between the front-end and back-end work. The back-end HRA analysis description cited the use of "EPRI Operator Reliability Experiment data" without any reference.

The back-end HRA analysis description stated that the HEP for initiation of Suppression Pool Cooling from an unidentified PRA was used in the IPE. This operation is plant and event sequence specific. Information regarding the HEP from this unidentified PRA would be necessary to assess tha appropriateness of this HEP.

No numerical screening of HEPs was performed. Important operator functions were identified by constructing Event Sequence Diagrams during the accident delineation and system analysis (mainly from EOPs and supporting procedures) and estimated HEP values were input into the Event trees. The choice of actions to be analyzed is a form of screening. Incorporation of operator actions into the PRA is discussed in Section 2.3.1.

Pre-initiator events (e.g. maintenance errors, including testing, equipment calibration, and restoration) were addressed in the IPE. A qualitative screening was performed using various criteria described below. Industry and plant specific data (including procedures) and the

qualitative screening criteria were used to conclude that only SLS and ECCS required evaluation.

Qualitative screening criteria included: (1) single train with independent I&C (only common cause human induced errors are of concern); (2) systems monitored and alarmed in the control room if misaligned; (3) systems with auto repositioning of misaligned valves; (4) system failures that could be caught by double verification during system restoration, post maintenance testing, or during routine operability checks.

Item 1 above is the criteria which screens most systems from the pre-initiator human error making systems inoperable. The submittal, in Section 3.3.3.4, contained the following justification for this assumption:

> "An important assumption implicit in this analysis is that human induced failure modes affecting single train systems, caused by a maintenance related error, are subsumed by estimates of equipment unavailability due to test and maintenance activities and equipment failure rates. This assumption appears to be supported by industry and plant specific data."

A discussion addressing the efforts by the licensee to ensure the application of this assumption to NMP2 would be necessary to assess the appropriateness of this assumption.

2.1.3 WR 1.1.3 The methodology (including the human action taxonomy) employed is capable of identifying important human actions, and contains a discussion of the most important human actions and errors.

The process used by NMP2 analysts to identify the important human actions started with identification of potential accident intitiators. Initiators were identified from examination of five other PRAs noted in the submittal, from WASH-1400, from the list provided in NUREG/CR-3862, and from plant specific information such as LERs. Then the plant response (automatic actions) and EOPs were reviewed to identify the event sequences. Within those event sequences the significant operator functions to be accomplished, e.g., depressurize the reactor pressure vessel, were identified. The operator functions were broken down to identify basic HIs for quantitative analysis (HEP estimates). The submittal provides no discussion of the process for breakdown and analysis of the operator functions into basic HIs. It does provide (Section 3.3.3.2) some examples of the results of the breakdown, including task cues, procedures, time available, potential dependencies among individual human actions, and other factors influencing success or failure. The submittal would have been strengthened if the process for obtaining this information were described.

The submittal implies that detailed information comparable to the examples shown in Section 3.3.3.2 exists for all HIs analyzed. However, discussion of operator actions in the submittal, except for those examples, is at a higher level. For example, operator actions included in the Top Events are discussed, and information such as time constraints to performance of those actions is provided.

As a check on the results of the identification process, the HIs selected for analysis at NMP2 was compared to the list analyzed in the Grand Gulf PRA (Ref. 5). The NMP2 IPE considered those analyzed for Grand Gulf plus additional EOP actions specified for NMP2.

Based on the general explanation of the overall process for identification of important human actions, the sample results provided in the submittal, and on the comparison with the Grand Gulf PRA, it appears that the process used to identify important human actions was reasonable. **However, the discussion of the process for breaking down and analyzing operator functions to identify basic human interactions for quantitative analysis was not well documented.**

2.1.4 WR 1.1.4 The IPE submittal employed a viable process to confirm that the IPE represents the as-built, as operated plant.

Information assembly is discussed in Section 2.4 of the submittal. Plant documentation and plant walkdowns were a very important part of the process which was put into place to ensure the IPE process represents as-built, as-operated plant. The plant documentation included EOPs, plant operating procedures, and other procedures. Plant drawings and FSAR, Design Basis Documents, LERs, test results and maintenance work reports were also used.

The HRA analysts were involved in the system walkdowns which included interviews and discussions with operators and training personnel, and simulator observations and exercises. Details of the interview process and simulator observations were not provided.

In addition to documentation and walkdowns, an IPE Support Team made up of training personnel, plant operators, EOP development personnel and system engineers provided support to the IPE analysts. Internal reviews by a cross-section of engineers, operators, and trainer were a part of the process. The internal review team was led by QA and Independent Safety Engineering Group personnel.

With the exception of the lack of information idnetified above, the submittal provides a reasonably comprehensive description of the process for confirming that the IPE represents the as-built, as-operated plant; and, that process appears to have been an effective one.

2.1.5 WR 1.1.5 The HRA had been peer-reviewed to help assure the analytic techniques were correctly applied

An independent review team reviewed the IPE documentation. The team was headed by the QA organization and the Independent Safety Engineering Group (ISEG). Responsibility for the HRA review was divided between QA, ISEG and Operations reviewers. The qualifications of the HRA reviewers was not detailed in the submittal.

2.2 Work Requirement 1.2 Review the most likely sequences that could occur at the plant.

2.2.1 WR 1.2.1 The accident sequences appropriately considered human actions consistent with other NUREG-1150 and other NRC accepted PSAs (see table NUREG-1335 Appendix B).

The human actions considered in this IPE were compared to the Grand Gulf PRA (Ref. 5). The HIs evaluated in the Grand Gulf PRA are included in the NMP2 IPE plus additional NMP2 EOP actions and recovery actions. The HEP values for comparable HIs analyzed were consistent with the Grand Gulf HRA values.

2.2.2 WR 1.2.2 The accident sequences screened out because of low human error (see NUREG-1135, Section 2.1.6.6) appears appropriate, based on HRA techniques employed.

Important Split Fractions which include human actions for core damage frequency (CDF) are discussed in Section 3.4.2.5 of the submittal. The importance and risk reduction worth for top event split fractions containing human actions was provided. Many of the split fractions also contained equipment failures, so the contribution of human actions cannot be ascertained from information provided. While a list of important operator functions was provided, no discussion of HIs which are important to event sequences or CDF was provided. This omission, along with lack of details for HEP evaluations make it difficult to evaluate HI effect on CDF.

2.3 Work Requirement 1.3 Review the quantitative nature of the IPE submittal.

2.3.1 WR 1.3.1 The employed human error probability (HEP) screening values appear capable of screening <u>in</u> significant human errors.

No numerical screening of HEPs typical of many PRAs was identified from the submittal review. Important operator functions were identified by constructing Event Sequence Diagrams during the accident delineation and system analysis (mainly from EOPs and supporting procedures) and evaluated HEP values were input into the event trees. The choice of actions to be analyzed is a form of qualitative screening. It is assumed that the Tier 2 documentation will provide details of the decision process and selection criteria for actions that received quantitative analysis. In addition, pre-initiator human errors were also screened by use of selection criteria described in Section 2.1.2 of this TER. The submittal would be strengthened if criteria and process for qualitative screening were described.

2.3.2 WR 1.3.2 The IPE developed human error probabilities (HEPs) for significant human actions, or provided rationale for using screening values.

Actions selected for analysis were analyzed directly, and HEPs were developed. Given proper justification for selection of the actions to be analyzed, use of "best estimate" HEPs rather than screening values should not be viewed as a weakness. Selection of operator actions to be evaluated is discussed in Section 2.3.1 above.

2.3.3 WR 1.3.3 Sources of generic human reliability data used in the IPE were identified and rationale for their use provided. Generic human error probabilities (HEP) data

were modified using plant-specific Performance Shaping Factors (PSFs) as appropriate, and rationale provided for selection of employed PSFs.

The submittal discusses at a high level usage of information gathered on plant-specific conditions, limitations of personnel, specific cues, procedural guidance available, etc., that was used to evaluate the HEPs for operator actions. Several examples were provided which used THERP and ASEP sources for data. Other than for the specific examples, the submittal did not provide details of the sources of generic data, or of the assumptions made, in applying plant-specific
PSFs. To evaluate this criteria additional information about input data and how plant-specific PSFs were used to modify generic data for each of the different methods used to analyze HIs is required.

2.3.4 WR 1.3.4 The recovery method is clearly described and credit for recovery actions appear justified.

Recovery actions are included in the system event trees and the top level event trees identified in procedures. System event trees include operator actions to recover system function after component failures. Operator intervention provided for in the EOPs is included in the top event Trees. Recovery actions are described as part of the system descriptions in section 3.2.1 of the submittal, and in the top level event tree discussions in section 3.1.2.

Table 3.3.3-1 of the submittal summarizes human error probabilities. The NMP2 values were compared to the Grand Gulf (Ref. 5) values. The recovery actions treated are actions for which guidance is provided in the EOPs. The HEP for one human action, "Align containment heat removal (OH1)" for non-ATWS sequences, has a value of 1.0E-5, which appears to be low. The HEP values for the other actions appear to be reasonable and comparable to the Grand Gulf values for similar proceduralized actions.

2.4 WR 1.4 Review the IPE approach to reducing the probability of core damage or fission product release.

2.4.1 WR 1.4.1 The IPE analysis appears to support the licensee's definition of vulnerability, and that the definition provides a means by which the identification of potential vulnerabilities (as so defined) and plant modifications (safety enhancements) is made possible.

Section 3.4.2 of the IPE submittal compares the results to the proposed safety goals for core damage frequency of 1.0 E-4 and concludes that there are no vulnerabilities for NMP2. The submittal did not describe a screening process for improvements. Important sequences, top event split fractions and human action split fractions which that contributors to core damage frequency were identified and ranked in Section 3.4 of the submittal. Section 6 discusses modifications and improvements being implemented or considered as a result of the review of the major contributors to core damage. Section 4.8 discusses Accident Management insights for severe accidents provided as a result of this IPE. Despite the lack of a clear description in the IPE, and disregarding all comments about the HRA methodology, the overall process employed

in the IPE appears to have systematically identified "insights" as intended by the Generic Letter. Section
2.4.2 below provides a summary of the results of the identification of cost effective insights.

2.4.2 WR 1.4.2 The identification of plant improvements include human-related plant modifications (e.g., procedures and training,), and proposed modifications are reasonably expected to enhance human reliability and plant safety.

Cost beneficial plant improvements identified during the IPE process and incorporated are discussed in Section 6.2 of the submittal. Procedural guidance was added for several operator actions identified as important to mitigating plant conditions: opening purge valves for different combinations of loss of support systems of instrument air or nitrogen supply or electrical power supply for containment venting; opening doors to ECCS pump rooms with loss of room cooling; Station Blackout procedures (from Station Blackout initiative); procedural guidance to open outside doors for Service Water or Fire Water flooding; and addition of precautions to test and maintenance procedures to ensure no inadvertent opening of low pressure injection paths during power operation (reduce probability of interfacing system LOCA). No information was provided on the improvement in the IPE results.

In addition, Section 6.3 of the submittal provides improvements which are under consideration. Many of the items are related to Station Blackout procedures and EOPs. These items will require additional technical evaluation as they deal with use of these procedures in severe accidents. Also, Accident Management insights were provided from the Level 2 evaluation. Incorporation of these suggested improvements provide procedural guidance for many severe accident sequences. This is expected to enhance performance of plant personnel under these conditions.

3. OVERALL EVALUATION AND CONCLUSIONS

With regard to the HRA, the submittal suggests that the licensee used a reasonable process to meet the intent of Generic Letter 88-20. Overall, the HRA methodology used for identification of important actions, analysis of factors influencing human performance, quantification of human error, assessing the impact of human error on system response (and therefore CDF and releases) appears reasonable and consistent with practice in other PSAs. A reasonable process was in place to identify potential human-related improvements.

However, a notable weakness of the submittal is the absence of important information on the implementation of all three methodologies used as the primary tools for quantification of human error. This information includes the rationale for selection of a particular method for particular human actions or types of human actions, and sufficient discussion of the inputs, assumptions and rationale employed by the HRA analysts, so that an external reviewer, or a subsequent analyst, can trace the analysis and understand the technical basis for estimated HEPs.

REFERENCES

1. Swain and Guttman, "Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications," NUREG/CR-1287, August 1983.

2 USNRC, "PRA Procedures Guide," NUREG/CR-2300, January 1983.

3. EPRI, "An approach to the Analysis of Operator Actions in Probabilistic Risk Assessment," EPRI-TR-100259, December 1991.

4. Swain, "Accident Sequence Evaluation Program Human Reliability Analysis Procedure," NUREG/CR-4772, February 1987.

5. USNRC, "Analysis of Core Damage Frequencies from Internal Events: Grand Gulf-1," NUREG/CR-4550/Vol. 6.

ENCLOSURE 5

SUMMARY OF THE NINE MILE POINT NUCLEAR STATION, UNIT 2

INDIVIDUAL PLANT EXAMINATION (IPE)

SUBMITTAL ON INTERNAL EVENTS

Summary of the Nine Mile Point Nuclear Station, Unit 2 Individual Plant Examination (IPE) Submittal on Internal Events

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. The licensee's IPE is based on a full scope level 2 PRA performed in fulfillment of GL 88-20 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 core damage frequency.

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. As a result of this review, the licensee has 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. The licensee probed the results by performing importance analyses for top events, split fractions, and operator actions. 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."

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.

The licensee's IPE results(*) are summarized below.

Plant Type: BWR 5

Containment type: Mark II

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 (non-vented)	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

  - 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.)

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,
Original signed by:
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

Distribution: *w/enclosures 1-5

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