BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2,

INDIVIDUAL PLANT EXAMINATION

TECHNICAL EVALUATION REPORT

(HUMAN RELIABILITY ANALYSIS)

Enclosure 4

CONCORD ASSOCIATES, INC.
Systems Performance Engineers

CA/TR 93-019-29

BRUNSWICK STEAM ELECTRIC PLANT
UNITS 1 AND 2

TECHNICAL EVALUATION REPORT ON THE IPE SUBMITTAL HUMAN RELIABILITY ANALYSIS

FINAL REPORT

By

P.M. Haas

Prepared for:

U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Division of Safety Issue Resolution

Draft Report July, 1994
Final Report May, 1995

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E. EXECUTIVE SUMMARY

This Technical Evaluation Report (TER) is a summary of the documentation-only review of the human reliability analysis (HRA) presented as part of the Carolina Power and Light Company (CP&L) Brunswick Steam Electric Plant (BSEP) Units 1 and 2 Individual Plant Examination (IPE) submittal to the U.S. Nuclear Regulatory Commission (NRC). The review was performed to assist NRC staff in their evaluation of the IPE and conclusions regarding whether the submittal meets the intent of Generic Letter 88-20.

E.1 Plant Characterization

BSEP is a dual-unit site with two BWR 4 reactors with Mark I containments. Commercial operation was initiated in 1975 and 1977, respectively, for Units 1 and 2. Rated power for each unit is 2436 MWt and 821 MWe (net). The front-end reviewers identified several risk-significant design features that involve human action. Several features that have positive impact (tend to reduce core damage frequency) noted were: (1) the ability to crosstie 1E buses between units, (2) the ability to vent containment, and (3) the ability to flood core/containment with service water or diesel driven firewater via the RHR system. One negative feature cited (tends to increase core damage frequency) is the need to stop venting before complete depressurization system. BSEP has a plant-specific training simulator. The BWR Owners Group Rev. 4 Emergency Procedure Guidelines have been implemented.

E.2 Licensee IPE Process

The BSEP HRA was performed by CP&L personnel with additional expertise provided by Halliburton NUS Corporation. Significant utility staff involvement, plant walkdowns and document review helped to assure that the IPE/HRA represented the as-built, as-operated plant. Independent review helped to assure appropriate application of the technical approaches employed. The HRA process addressed both pre-initiator and post-initiator actions. Pre-initiator actions considered included both restoration errors and miscalibration. Post-initiator actions included both response-type and recovery-type actions. Pre-initiator human errors were quantified using a single value for calibration errors and a single value for restoration errors. The two values were based on a THERP (Ref. 1) calculation with some plant-specific considerations. Post-initiator human errors were quantified using primarily an EPRI approach (Ref. 2), supported by other techniques/approaches, including the Risk Methods Integration and Evaluation Program (RMIEP) HRA methodology (Ref. 3), and the Daugherty and Fragola approach (Ref. 4) Plant-specific performance shaping factors and dependencies were considered in the analysis of post-initiator human errors.

E.3 Human Reliability Analysis

E.3.1 Pre-Initiator Human Actions.

The BSEP HRA addressed pre-initiator errors in maintenance, test and surveillance actions by incorporating human error into the systems analysis (fault trees) as a specific cause for system unavailability. Both misalignment (restoration errors) and miscalibration were considered. Pre-initiator actions to be quantified were identified and selected by the PRA analysts from operating procedures and functional test procedures during the development of the system models and failure sequences. Some actions were removed from further consideration by application of "qualitative screening" guidelines. Our review indicated that those guidelines were reasonable and consistent with practice in accepted PRAs. No numerical screening process was employed to eliminate pre-initiator human errors from detailed quantification. Forty three pre-initiator human errors were included in the IPE model, 10 calibration errors, and 33 restoration errors.

The approach used to quantify pre-initiators is somewhat unusual in comparison to other IPEs reviewed to date. A single human error probability (HEP) value (4.17E-03) was used for all calibration errors, and another single HEP value (9.5E-03) was used for all restoration errors. The value in each case was determined from an analysis using a THERP tree representing a typical error. One tree represented all calibration errors; the other tree, all restoration errors. The THERP analysis considered some global performance shaping factors related to BSEP maintenance practice, e.g., the use of two-person teams and the use of separate procedures for each calibration. Situation-specific or equipment-specific performance shaping factors such as the human interface design, environmental conditions, access, quality of procedures, etc. were not addressed. Thus, the HEP values can be considered essentially "generic" values, even though some plant-specific features were considered in the THERP calculations. Each pre-initiator HEP was multiplied by a factor intended to represent the fraction of operating time that the system may be unavailable due to the pre-initiator error. This fraction is the ratio of the estimated time until detection to the time of operation (i.e., the interval between testing).

E.3.2 Post-Initiator Human Actions

The BSEP HRA addressed both response-type and recovery-type post-initiator human actions. Two types of response actions were modeled: 1) actions used to manually start or align components after a failure of automatic actuation (modeled in system fault trees), and 2) high level actions (in event tree headings) involving system realignment to restore failed success paths to mitigate the consequences of an accident sequence. Actions both inside and outside of the control room were modeled. All response actions modeled were proceduralized. Eleven recovery actions were modeled, including three related to restoration of offsite power within a specified time. Most, though not all, recovery actions credited were proceduralized.

No numerical screening process was employed to eliminate operator actions from the more detailed quantification. HEPs were developed for all of the operator actions identified. As indicated above, several HRA techniques and data sources were used to quantify post-initiator human errors, the primary one being the method in EPRI NP-6560-L. Each response action is considered as a combination of two types of actions: 1) detection/ diagnosis/decision, or "cognitive" action, and 2) manual action. Errors can occur in the cognitive action via failures in cognitive processing or procedural "mistakes", or they can occur by failing to process information in a timely manner. Errors in manual actions are considered manipulative "slips". The total HEP is a probabilistic combination of the probabilities of failure by each of the three mechanisms:

- P_1 - mistakes in cognitive processing
- P_2 - failure to process information in time
- P_3 - slips.

A range of generic values are provided for P_1 and P_3, along with guidance for the analyst to subjectively evaluate the plant-specific situation and select one of the screening values. The value for P_2 is calculated from a simplified "time reliability correlation" which provides HEPs as a function of the ratio of time required to time available for the operator action. Time available was determined from transient analysis (thermal hydraulic) codes. Required time was based on the judgment of analysts and operators, with some general input from observed simulator exercises. Plant-specific performance shaping factors and dependencies were considered in the analysis. Comparison of the HEP values with values from other PRAs for similar actions indicates that, in general, the BSEP estimates are consistent with typical values used in accepted PRAs (NUREG-1150) and other BWR IPEs.

E.4 Generic Issues and CPI

The licensee's consideration of generic safety issues (GSIs) and unresolved safety issues (USIs) and of containment performance improvements (CPI) recommendations are the subject of the front-end review, and back-end review, respectively. The licensee addressed some aspects of decay heat removal (DHR), and indicates that unresolved safety issue (USI) A-17 is addressed through the analysis of internal flooding. Operator actions associated with DHR are identified as important actions and are discussed in the submittal. The back-end reviewer noted that the licensee concluded that with planned modifications associated with a hardened wetwell vent, no further changes were required to resolve the Mark-I containment issues. These modifications were not credited in the IPE.

E.5 Vulnerabilities and Plant Improvements

The submittal does not provide a precise definition of a vulnerability. The NUMARC Severe Accident Closure Guidelines (NUMARC 91-04) were used to develop criteria to determine the appropriate action to take in response to issues identified from the IPE analysis - reanalysis, plant modifications, procedure changes, severe accident management guidance, or no action. As indicated above, the licensee's assessment included review of plant modifications already committed for installation which will address some of the issues identified in the IPE, in particular, the contribution from operator actions noted above for the Station Blackout and Loss of Decay Heat Removal sequences. These enhancements included:

- Installation of a system to facilitate remote operation of the inter-unit emergency bus cross-tie
- Installation of a hardened wetwell vent
- Development of new procedures for DC power recovery and station blackout events.

E.6 Observations

The following observations from our review are pertinent to NRC's decision regarding whether the licensee met the intent of GL 88-20:

(1) The submittal and supporting documentation indicates that utility personnel were involved in the HRA, and that the walkdowns and documentation reviews constituted a viable process for confirming that the HRA portions of the IPE represent the as-built, as-operated plant.

(2) The licensee performed an independent review that provides some assurance that the HRA techniques have been correctly applied and that the documentation is accurate.

(3) Pre-initiator human errors, both restoration errors and miscalibration, were considered in the analysis. The approach for identification and selection of pre-initiator actions included review of procedures and discussion with plant personnel.

(4) The quantification process for pre-initiator human actions involved some, but relatively limited, consideration of plant-specific factors that could influence human error probability. Thus the analysis provides limited insight as to plant-specific influences on these potential contributors to plant risk. It is a positive finding that numerical results generally were consistent with other PRAs, and that pre-initiator human actions were identified as some of the important human actions. However, a more in-depth assessment of plant-specific procedures and practice related to routine operations,

maintenance, test, calibration, surveillance, etc. would provide the licensee with an enhanced understanding of actual performance at BSEP.

(5) The analysis of post-initiator human actions was reasonably complete in scope. Both response-type and recovery-type actions were included. The process for identification and selection of actions to be quantified included review of procedures and discussion with plant personnel. The quantification process considered timing of operator actions and, to a limited degree, other plant-specific performance shaping factors that could influence human error probability. The understanding of plant-specific influences on human behavior is limited by the degree of in-depth plant-specific assessment.

(6) Dependency among multiple human actions was addressed by the licensee, but the discussions by the licensee suggest that the analysis was limited in depth and rigor. This issue may be simply a problem of lack of thorough discussion/documentation; or it could be a weakness of the licensee's methodology. Failure to account for dependencies could lead to overly optimistic estimates of the impact of human error in response to accident sequences. Individual HEPs were, in general, consistent with results in other PRAs; but it is not possible from this document-only review to assess the impact of dependency assumptions on the overall IPE results.

(7) The licensee employed a reasonable screening process for identifying vulnerabilities. Enhancements already planned were identified as providing significant reduction in the estimated core damage frequency. No additional human-performance-related enhancements resulting specifically from the IPE were identified by the licensee.

I. INTRODUCTION

This Technical Evaluation Report (TER) is a summary of the documentation-only review of the human reliability analysis (HRA) presented as part of the Carolina Power and Light Company (CP&L) Brunswick Steam Electric Plant (BSEP) Units 1 and 2 Individual Plant Examination (IPE) submittal to the U.S. Nuclear Regulatory Commission (NRC). The review was performed to assist NRC staff in their evaluation of the IPE and conclusions regarding whether the submittal meets the intent of Generic Letter 88-20.

1.1 HRA Review Process

The HRA review was a "document-only" process which consisted of essentially four steps:

(1) Comprehensive review of the IPE submittal focusing on all information pertinent to HRA.

(2) Preparation of a draft TER summarizing preliminary findings and conclusions, noting specific issues for which additional information was required from the licensee, and formulating requests to the licensee for the necessary additional information.

(3) Review of preliminary findings, conclusions and proposed requests for additional information (RAIs) with NRC staff and with "front-end" and "back-end" reviewers

(4) Review of licensee responses to the NRC requests for additional information, and preparation of this final TER modifying the draft to incorporate results of the additional information provided by the licensee and finalize conclusions.

Findings and conclusions are limited to those that could be supported by the document-only review. No visit to the site was conducted. No discussions were held with plant personnel or IPE/HRA analysts, either during the initial review of the submittal or after receipt of licensee responses to NRC RAIs. No review of detailed "Tier 2" information was performed, except for selected details provided by the licensee in direct response to NRC RAIs. In general it was not possible, and it was not the intent of the review, to reproduce results or verify in detail the licensee's HRA quantification process. The review addressed the reasonableness of the overall approach with regard to its ability to permit the licensee to meet the goals of Generic Letter 88-20.

1.2 Plant Characterization

BSEP is a dual-unit site with two BWR 4 reactors with Mark I containments. Commercial operation was initiated in 1975 and 1977, respectively, for Units 1 and 2. Rated power for each unit is 2436 MWt and 821 MWe (net). The front-end reviewers identified several risk-significant design features that involve human action. Several features that have positive impact (tend to reduce core damage frequency) noted were: (1) the ability to crosstie 1E buses between units, (2) the ability to vent containment, and (3) the ability to flood core/containment with service water or diesel driven firewater via the RHR system. One negative feature cited (tends to increase core damage frequency) is the need to stop venting before complete depressurization system. BSEP has a plant-specific training simulator. The BWR Owners Group Rev. 4 Emergency Procedure Guidelines have been implemented.

II. TECHNICAL REVIEW

2.1 Licensee IPE Process

2.1.1 Completeness and Methodology

The submittal information on the HRA process was generally complete in scope. Several areas were lacking in detail or unclear. Additional information obtained from the licensee in response to the NRC RAIs was sufficient to complete our assessment of the overall HRA.

The HRA process addressed both pre-initiator and post-initiator actions. Pre-initiator actions considered included both restoration errors and miscalibration. Post-initiator actions included both response-type and recovery-type actions. Pre-initiator human errors were quantified using a single value for calibration errors and a single value for restoration errors. The two values were based on THERP (Ref. 1) calculations which included consideration of some global plant-specific performance shaping factors. Post-initiator human errors were quantified using primarily an EPRI approach (Ref. 2), supported by other techniques/approaches, including the RMIEP HRA methodology (Ref. 3) and the Daugherty and Fragola approach (Ref. 4). Plant-specific performance shaping factors and dependencies were considered in the analysis of post-initiator human errors.

2.1.2 Multi-Unit Effects and As-Built, As-Operated Status

The IPE model is based on Unit 2. The submittal indicates that potential differences between the units were considered and that there are not significant differences for Unit 1 that necessitate a separate model. The two units share the control room, intake structure, electrical power systems (including diesel generators and 1E buses), and control room HVAC. The issue of dual unit core damage is addressed by the front-end reviewer.

The freeze date for the IPE was January 1, 1992. Since the IPE was based on the earlier probabilistic risk assessment (PRA), a "complete reverification" of the PRA model was performed by the Corporate Nuclear Safety Section with assistance from NUS. Included in this reverification were:

- A comparison of the PRA model descriptions with the plant systems descriptions and systems diagrams to confirm that the PRA documentation was consistent with the current plant design

- Review of NUREG/CR-5465, "NRC Review of the Brunswick PRA," and verification that each of the issues raised by the NRC contractor review had been adequately disposed in the IPE models

- Review of assumptions used in the development of event and fault trees to assure that they were still valid or to modify them to reflect current information

- Redevelopment of the ATWS and Station Blackout Event Trees "to take advantage of advances in PRA methodology, and changes made to the plant design and plant procedures" since the initial PRA

- Independent review by plant operations and training staff, plant engineering and licensing staff, and by consultants to assure that the IPE reflected a complete and up-to-date picture of the as-built, as-operated plant. (See Section 2.1.2 below.)

Plant systems had recently been walked down by systems engineering staff as part of the design basis reconstitution activities. Therefore, the IPE team assumed that the information in the plant P&IDs reflected the as-built configuration, and did not conduct general, broad-scoped plant walkdowns. Walkdowns were performed in support of specific issues or IPE analyses, e.g., in the flooding analysis and Level 2 analysis. No plant walkdowns focused specifically on HRA issues were noted in the submittal. However, accident sequences were talked through with operators and members of the training staff to verify that operator actions were guided by procedure, that they were feasible under the conditions expected during the accident, that they were practiced on the simulator, and that they could be performed within the available time. Some scenarios were presented as simulator exercises and observed by IPE team members.

These various actions taken by the IPE team to verify and build on the documentation provided from the original PRA, the involvement of CP&L personnel with plant engineering and operations knowledge, and independent reviews by engineering, operations and training staff appear to have provided reasonable assurance that the IPE represents the as-built, as-operated plant as of the identified freeze date.

2.1.3 Licensee Participation and Peer Review

2.1.3.1 Licensee Participation. The submittal notes that CP&L development of PRA expertise began in the early 1980s. A Level 1 PRA was completed for BSEP in 1987. That project was co-managed by CP&L. Technical direction and approximately half of the technical effort was performed by a contractor, with technology transfer to CP&L as a major objective. This initial PRA served as the foundation for the IPE effort, which was performed from late 1989 through mid-1992 by the corporate PRA group with technical assistance from Halliburton NUS, Inc., Risk Management Associates, and SAROS, Inc.

The corporate organization responsible for the IPE is the Risk Assessment Section in the Nuclear Engineering Department. The section is organized into three groups - a PWR unit, a BWR unit, and a "PRA Disciplines" unit, which provides technical support for both the PWR and BWR units. The HRA members of the IPE team are in the PRA Disciplines unit. Two CP&L persons are identified as HRA analysts, one of whom appears to have operations expertise. One NUS contractor who has HRA expertise, Dr. Gareth Parry, is also identified as a team member under the PRA Disciplines unit, though the submittal does not state directly that he was involved with the HRA. The submittal notes plant design and operations knowledge/experience on the part of other members of the IPE team. Most pertinent to the HRA, the submittal notes that operations training staff provided information to the HRA analysts regarding time required to recognize the accident condition, the time required to perform response actions, the time window available to complete the action, and the environment faced by the operator (competing actions and demands, adequacy of procedures and training, etc.) Operator training staff also assisted with simulator exercises conducted to verify selected HRA input.

2.1.3.2 Peer Review. The submittal (Section 5.2) discusses the following independent reviews of the IPE analysis and results:

- The review of the initial BSEP PRA by NRC staff and contractors from Idaho National Engineering Laboratory (INEL), published in NUREG/CR-5465. Comments and findings from this review were addressed in the development of the IPE.

- Independent internal reviews by plant operations training personnel. PRA staff reviewed event trees with a member of the operations training staff who had plant operating experience (Senior Reactor Operator) and experience in training and development of EOPs. Operations training staff also reviewed the results of timing studies performed with a thermal hydraulics code (MARCHRMA) to verify consistency with assumptions in simulator training.

- Reviews by Plant Systems Engineers. Cognizant systems engineers and operations staff reviewed the IPE systems modeling (fault trees) both at the time of the original PRA and at the time of development of the IPE.

- Review by Consultants. NUS staff reviewed event trees and their supporting assumptions and timing studies, calculations supporting the flooding analyses, many of the significant fault trees, and some of the HRA results.

No specific review comments or resolutions from these reviews were discussed in the submittal. In our opinion, these reviews collectively constituted a reasonable process for an "in-house" peer review that provides some assurance that the IPE analytic techniques were correctly applied and that documentation is accurate.

2.2 Pre-Initiator Human Actions

Errors in performance of pre-initiator actions (i.e., actions performed during maintenance, testing, etc.) may cause components, trains, or entire systems to be unavailable on demand during an accident, and thus may significantly impact plant risk. Our review of the HRA portion of the IPE examines the licensee's HRA process to determine what consideration was given to pre-initiator human actions, how potential errors were identified, the effectiveness of quantitative and/or qualitative screening process(es) employed, and the processes for accounting for plant-specific performance shaping factors, recovery factors, and dependencies among multiple actions.

2.2.1 Pre-Initiator Human Actions Considered.

The BSEP HRA addressed pre-initiator errors in maintenance, test and surveillance actions by incorporating human error into the systems analysis (fault trees) as a specific cause for system unavailability. Both misalignment (restoration) errors and miscalibration were considered.

2.2.2 Process for Identification and Selection of Pre-Initiator Human Actions.

The key concerns of our review regarding the process for identification and selection of pre-initiator human actions are: (a) whether maintenance, test and calibration procedures for the systems and components modeled were reviewed by the systems analyst(s), and (b) whether discussions were held with appropriate plant personnel (e.g., maintenance, training, operations) on the interpretation and implementation of the plant's test, maintenance and calibration procedures to identify and understand the specific actions and the specific components manipulated when performing the maintenance, test, or calibration tasks.

The submittal indicates that BSEP PRA procedures related to system analysis and development of fault tree models specifically call for identification and inclusion of human errors in calibration and in restoring equipment after test and maintenance. The submittal also states (page 3.3.14) that plant-specific calibration and restoration procedures and practices were examined to be sure that the HRA models were realistically developed and quantified. Further, guidance for the analysis of dependencies (common cause) specifically called for identification of system dependencies induced by human error.

Restoration errors were not modeled for components which:

- automatically realign upon demand
- are tested when the maintenance activity is completed
- are unaffected by maintenance
- annunciate or would be identified by checks performed either once a shift or once a day.

These guidelines, or qualitative screening rules, are reasonable and are generally consistent with guidance in accepted methods and with practice in other accepted PRAs. Obviously, it is important for the analyst to verify that the modeling assumptions represent actual plant-specific practice and procedures. As discussed in Section 2.2.4 below, the submittal indicates that this confirmation was obtained and lists some of the specific practices and procedures that apparently were verified.

2.2.3 Screening Process for Pre-Initiator Human Actions.

No numerical screening process was employed to eliminate pre-initiator human errors from detailed quantification. All pre-initiator errors were assigned an HEP incorporating a value based on a THERP (Ref. 1) calculation plus assumptions about the likelihood of detection of error during operation before an accident occurs. The quantification approach is discussed below in Section 2.2.4. Forty three pre-initiator human errors were included in the IPE model; 10 calibration errors, and 33 restoration errors.

2.2.4 Plant-Specific Performance Shaping Factors, Recovery Factors, and Dependencies for Pre-Initiator Human Actions.

The BSEP approach used to quantify pre-initiators is somewhat unusual in comparison to other IPEs reviewed to date. It did include some plant-specific consideration of maintenance, test, and calibration procedures and practice, yet it resulted in application of a single HEP value for all calibration errors (HEP=4.17E-03) and another single HEP value for all restoration errors (HEP=9.5E-03). The value in each case was determined from an analysis using a THERP tree representing a typical error. One tree represented all calibration errors; the other, all restoration errors. Thus performance shaping factors specific to the particular human-equipment interface, environmental conditions, quality of procedures, etc. were not addressed. Other more global factors, such as use of two-person teams, use of separate procedures for each calibration, etc. were considered in the THERP analysis. The plant-specific considerations employed in the THERP analysis for calibration and restoration errors are listed in Tables 2-1 and 2-2, respectively.

The two THERP trees are illustrated in the submittal, and the source of the nominal HEPs employed in the calculation (i.e., the specific THERP tables and values) are clearly documented. The assumptions made in selecting the THERP tables and applying specific THERP values appear to be consistent with the stated plant practice (as summarized in Tables 2-1 and 2-1), and the THERP calculation was performed correctly. The submittal states (page 3.3.14) that, "Plant-specific calibration and restoration procedures and practices were examined to be sure that the HRA models were realistically developed and quantified for BSEP."

Table 2-1
Plant Practice and Procedures Considered in the Quantification of Calibration Errors

1. The calibration program or the technical specifications direct equipment-specific calibrations to be performed every 3,6,12, or 18 months.

2. Each calibration is covered by a separate procedure sheet.

3. Calibration teams normally involve at least two people: a) one to perform the calibration; and b) the other to observe the work and check off each procedural step as it is completed.

4. Procedure sheets have "as found" and "as left" entries which are compared after the calibration is complete and before the calibration sheets are signed off by the shift foreman.

5. Generally, the I&C maintenance foreman checks the consistency between the "as found" and "as left" readings within three working days.

6. Some of the instrument panel readings are taken and recorded by reactor operators who observe and compare them with other readings during instrument loop calibration.

7. Calibration procedures are prepared by one person and checked by a second person or group.

8. I&C technicians can operate most instrument sensing line valves after obtaining approval to perform the test from the shift foreman. Other valves must be operated by an operator.

Table 2-2
Plant Practice and Procedures Considered in the Quantification of Restoration Errors

1. Maintenance can be performed during power operation if allowed by the Technical Specification guidelines.

2. Maintenance activities are generally directed by an applicable set of procedures.

3. Operating personnel perform all isolation and system realignments needed to perform maintenance.

4. The shift foreman generally checks the clearance tag sheet within eight hours of the time that maintenance is complete.

5. Maintenance teams normally involve at least two people: a) one person (or more) to perform the maintenance; and, b) one person to observe the work and check off each step as it is completed.

6. Normally, the operability of each component undergoing maintenance is confirmed when the work is complete. This may involve performance of applicable sections of plant surveillance test procedures.

The impact of human error in calibration or post-maintenance restoration is included in the IPE model through the evaluation of the unavailability of equipment on demand during an accident. The unavailability contribution due to these human errors, UA, is calculated as follows:

$$UA=\frac{(P)*(FDT)}{(IT)}$$

where: P = miscalibration (restoration) HEP
FDT = fault duration time before detection
IT = interval between calibrations (test or maintenance)

Pre-initiator human error frequently is incorporated into the PRA model by adjusting the component unavailability. And, the unavailability due to human error sometimes is adjusted to account for likelihood of detection (and recovery) of the error prior to the time that the demand (i.e., the accident) occurs. For the majority of the HEPs in the BSEP analysis, thefault duration time before detection is assumed to be one day. In other cases, when valves are locked open, the HEP is reduced by a factor of 10.

The licensee's response to an NRC RAI the licensee indicated that these assumptions were supported by review of applicable test and maintenance procedures, check lists, and plant standard practices, and discussion with knowledgeable operations staff. The licensee states that the one-day fault detection time is correct because the status of the components is verified each day, and a value of 0.99 is assumed as a recovery probability for the verification check. The assumed success rate of 0.99 may or may not be "realistic". Without access to specific procedures and a case-by-case assessment of the verification task, it is not possible for us to judge. The licensee assumes that the probability of failure on successive verification checks is independent, and therefore even if very conservative values of the failure probability were assumed, the probability of successive failure over a long period of time is extremely low. It may or may not be realistic to assume that the verification tasks truly are independent.

With regard to the assumption of multiplying by 0.1 the availability of valves required to be locked into position, the licensee explained that the rationale is that the act of locking the valve in position is considered as an independent verification of valve position. The value of 0.1 was assumed as a nonrecovery probability for the action of locking the valve into position. Again, without detailed assessment of the action, we are unable to judge whether this is a "realistic" nonrecovery probability or whether the locking of the valves is appropriately treated as an independent verification. It is apparent that the licensee is aware of the need to "validate" these assumptions made in the HRA; and the licensee's response indicates that these particular assumptions were verified by review of procedures and plant operational practice.

The submittal notes (Section 2.3.4.2) that the dependency analysis in the BSEP IPE included functional dependencies between safety and support systems, and between components within a single system. Four basic types of dependencies were included, one of which was "System dependencies induced by human actions." The submittal notes that, "These dependencies are typified by failures which result from operator, maintenance, and calibration errors." The licensee's response to an NRC RAI indicates that the licensee considered several aspects of dependencies in pre-initiator human actions.

Review of procedures and discussion with operations staff identified dependencies due to similar procedures and/or poor work environments. The licensee's response states that adjustments were made to results (HEPs) to account for such dependencies, though no details or specific examples are provided. The licensee's response also states that dependencies that could affect multiple trains were not considered. The basis provided for ignoring such potential dependencies is that maintenance on system trains is not conducted simultaneously, and the events are separated by time. The licensee notes, however, that subsequent to submittal of the IPE two potential pre-initiator actions were identified that could result in a common failure of instrument channels. The two actions are associated with calibration of the high drywell pressure detectors and reactor vessel low water level detectors. Calibration of these devices takes place during refueling. It is possible that a single crew could miscalibrate all four of the pressure detectors or all four of the level detectors. The licensee states that it is expected that the inclusion of these events would result in a negligible change in sequence frequencies and would not alter the results of the IPE submittal. No calculation or detailed discussion supporting this "expectation" is provided. There is not sufficient information presented for us to agree or disagree that the contribution is negligible. In some other PRAs, similar dependent failures have been significant contributors to core damage frequency. The discovery of this potential common failure associated with dependent human actions suggests that it indeed is prudent to examine plant-specific design and maintenance, test and calibration practice, and to not dismiss the possibility without thorough plant-specific evaluation.

It is our opinion that overall the HRA process employed for assessment of pre-initiator human actions was capable of providing the licensee with a reasonable understanding of the contribution to plant risk associated with human error in such actions. Strengths of the approach are that the licensee addressed both miscalibration and restoration errors, and that apparently assumptions regarding general BSEP practice and procedures in conducting maintenance and calibration that were made as part of the THERP analysis were verified by examination of procedures and discussion with plant personnel. A weakness is the use of a single HEP for calibration and a single HEP for restoration. A more in-depth, plant-specific and case-specific analysis provides opportunities for insights regarding specific factors that could reduce risk through enhanced human performance.

2.3 Post-Initiator Human Actions

Human error in responding to an accident initiator, e.g., by not recognizing and diagnosing the situation properly, or failure to perform required activities as directed by procedures, can have a significant effect on plant risk. These errors are referred to as post-initiator human errors. Our review determines the types of post-initiator errors considered by the licensee, and evaluates the processes used to identify and select, screen, and quantify post-initiator errors, including issues such as the means for evaluating timing, dependency among human actions, and other plant-specific performance shaping factors.

2.3.1 Types of Post-Initiator Human Actions Considered.

There are two important types of post-initiator actions considered in most nuclear plant PRAs: response actions, which include those human actions performed in response to the first level directives of the emergency operating procedures/instructions (EOPs, or EOIs); and, recovery actions, which include those performed to recover a specific failure or fault (primarily equipment failure/fault) such as recovery of offsite power or recovery of a front-line safety system that was unavailable on demand earlier in the event.

The BSEP HRA addressed both response and recovery actions. Two types of response actions were modeled: 1) actions used to manually start or align components after a failure of automatic actuation (modeled in system fault trees), and 2) high level actions (in event tree headings) involving system realignment to restore failed success paths to mitigate the consequences of an accident sequence. Actions both inside and outside of the control room were modeled, though the majority (67 out of 79) were in-control-room. All response actions modeled were proceduralized. Eight actions were modeled in event trees (seven in-control -room, and one ex-control-room); the remainder of the actions were modeled in fault trees.

Eleven recovery actions were modeled, including three related to restoration of offsite power within a specified time. Several different methods of quantification were used to develop HEPs for recovery actions. (Quantification of recovery actions is discussed in Section 2.3.5.3 of this TER.) The majority (at least 8 of the 11) of recovery actions are ex-control-room actions. The submittal does not clearly identify whether the recovery actions are directed by procedures. General discussion of the recovery analysis in Section 2.3.2.2 of the submittal implies that non-proceduralized actions were included. In response to an NRC RAI the licensee indicates that most human actions credited in the IPE are proceduralized, either in EOPs or other procedures, but that some actions "not strictly proceduralized" but "considered plausible" were included in the model. Actions that were not proceduralized were "reviewed by the operations staff to check their validity." The licensee's response further states that these actions were assigned a high failure probability and, therefore, their inclusion did not greatly affect the results. No specific examples of unproceduralized actions or associated HEPs were provided. Guidance in NUREG-1335 indicates that unproceduralized actions should not be credited without thorough assessment to assure that credit is appropriate. Without details of the analysis, it is not possible for us to judge the "realism" of the assumed

HEPs for particular cases. Based on the licensee's statements, it appears that few unproceduralized actions were credited, and that the licensee gave particular consideration to evaluation of those unproceduralized actions.

The method of incorporating recovery actions in the IPE model is not clearly specified in the submittal, and was not addressed in the NRC RAI. It appears that the eleven recovery actions identified in the submittal were added to cutsets after initial quantification. The discussion in Section 2.3.2.2 of the submittal indicates that recovery actions were included in both fault trees and event trees. Usually, it is desirable to incorporate recovery actions at the cutset level, because they are typically very cutset-dependent. It is possible that including recovery actions in fault trees or event trees could lead to an overestimate of the benefit (reduction in CDF) to be obtained because the recovery action may be credited where it does not apply. However, we have no evidence that this is or is not a significant issue with the BSEP IPE. The licensee may wish to confirm, or may already have confirmed that credit was not taken inappropriately due to the means of incorporating recovery actions into the IPE model.

2.3.2 Process for Identification and Selection of Post-Initiator Human Actions.

The primary thrust of our review related to this question is to assure that the process used by the licensee to identify and select post-initiator actions is systematic and thorough enough to provide reasonable assurance that important actions were not inappropriately precluded from examination. Key issues are whether: (1) the process included review of plant procedures (e.g., emergency operating procedures or system instructions) associated with the accident sequences delineated and the systems modeled; and, (2) discussions were held with appropriate plant personnel (e.g., operators or training staff) on the interpretation and implementation of plant procedures to identify and understand the specific actions and the specific components manipulated when responding to the accident sequences modeled.

The submittal does not provide much discussion of a systematic process for identification of human errors to be included in the IPE model. However, there are general statements in a number of discussions in the submittal indicating that procedures were reviewed and that operations and training personnel were appropriately involved in identification and review of operator actions. (Some of these were noted in Sections 2.1.1 and 2.1.2 above.) Systems analysts confirmed that all operator response actions credited were addressed in the emergency or system operating procedures. Table 3.3.3-10 of the submittal lists 79 operator errors that were included in the fault tree models, and Table 3.3.3-11 lists 8 that were included in the event trees. Comparison of these operator errors with those identified in other BWR PRAs indicates that the important operator actions typically addressed were included in the BSEP analysis. (See Table 2-5, Section 2.4.2 of this TER.)

2.3.3 Screening Process for Post-Initiator Response Actions.

The submittal does not discuss any numerical screening process that was employed to eliminate some operator actions from the more detailed quantification. HEPs were developed for all of the operator actions identified. However, in response to an NRC RAI on treatment of dependencies in post-initiators, the licensee does indicate that the initial model quantification was performed with all HEPs set at 0.1, and that the nominal values were then assigned to actions in cutsets remaining above the truncation limit. This sequence screening is discussed further in Section 2.3.6 below.

2.3.4 Quantification of Post-Initiator Human Actions.

The BSEP HRA employed a variety of different HRA techniques and data sources for quantification of different types of post-initiator human actions. The primary techniques were the EPRI methodology summarized in EPRI NP-6560-L (Ref. 2) and the approach presented by Daugherty and Fragola (Ref. 4). The approach to quantification and the consideration of timing and other performance shaping factors is summarized in the paragraphs below for each general type of operator action considered by the licensee.

2.3.4.1 Control Room Response Actions Modeled in Fault Trees. The submittal provides very little discussion of the EPRI technique. The EPRI report (Ref. 2) is not generally released to the public and was not available for this review. However, a later version of the model which is similar in its primary aspects is included in a later EPRI report (Ref. 5) which was available for review. Our discussion with one of the primary authors of the EPRI reports confirmed that the model described in Reference 5 is in all essential aspects the same as in Reference 2. The method was employed for in-control-room response actions that were modeled in the fault trees, which are the majority of the post-initiator actions modeled.

The submittal explanation of the EPRI model includes a graphic representation (Figure 3.3.3-1), which is reproduced as Figure 2.1 below. Each response action is considered as a combination of two types of actions: 1) Detection/diagnosis/decision, or "cognitive" action, and 2) manual action. Errors can occur in the cognitive action via failures in cognitive processing or procedural "mistakes", or they can occur by failing to process information in a timely manner. Errors in manual actions are considered manipulative "slips". The total HEP is a probabilistic combination of the three error probabilities P_1, P_2, and P_3 as illustrated in Figure 2.1.

In the BSEP analysis, screening values were used for P_1 and P_3. (Note that in this context the term "screening value" simply means a generic, presumably conservative, value that could be used for screening purposes. No numerical screening was performed. All of the HEPs quantified using these generic screening values were retained in the model.) While the screening values are generic, the selection of a particular screening value is guided by responses to questions which do address some of the key plant-specific and situation-specific performance shaping factors that would influence operator error probability. The screening values and the associated conditions for P_1 and P_3 are summarized in Table 2-3 below.

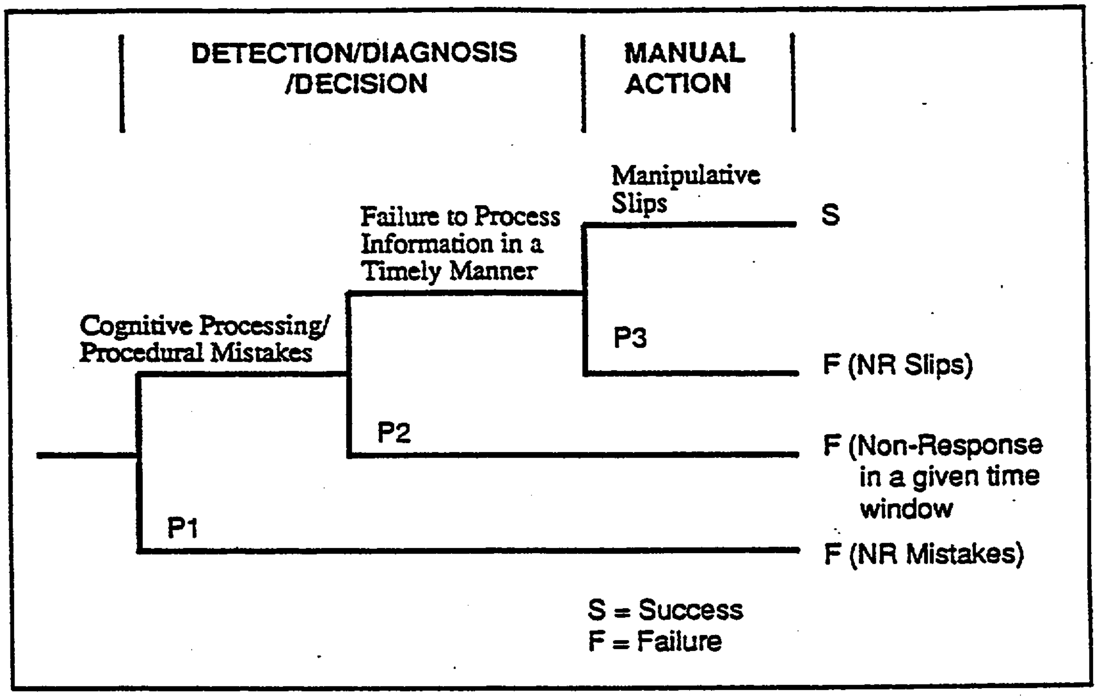

Figure 2.1 EPRI Model for Post-Initiator Response Actions

The value of P_2 was calculated from the lognormal function:

$$P_2 = 1 - \phi\left[\frac{\ln(T_w/T_{1/2})}{\sigma}\right]$$

where Tw = time window available
$T_{1/2}$ = time required for recognition
σ = logarithmic standard deviation.

This EPRI model and its basis have not received widespread peer review. It does appear to be a reasonable "notional" model. That is, it employs a systematic and logical process, but like other HRA techniques does not have a comprehensive theoretical or empirical basis. The "taxonomy" of human action/error types (i.e., each action consisting of a cognitive portion and a manipulative portion, and errors consisting of mistakes, slips, or failure to respond in time) is consistent with other currently accepted HRA approaches. The decision/event tree format with the criteria for selection of certain screening values provides guidance for the analyst to address some of the key performance shaping factors affecting the respective types of errors. The time correlation approach for estimating error probabilities for time-driven diagnostic/decision tasks is well recognized and has been widely used in HRAs conducted to date, and has some empirical basis over a limited range of conditions in simulator exercises.

Table 2-3
Screening Values and Selection Criteria for the EPRI Model

Parameter P_1 (Mistakes):

Screening Value	Conditions/Criteria
0.1 - 0.5	Known problems with parts of procedure; procedural routes not normally exposed during training; combination of time limited actions with little training; competing key actions; subtle actions disguised by well-known transients.
0.01	Unambiguous symptoms; slowly developing transient, but complex actions required; procedures somewhat unclear; limited training.
0.005	Unambiguous symptoms; slowly developing transient; simple actions but procedures somewhat unclear; limited training.
0.001	Unambiguous symptoms; clear procedures; limited training.
0.001	Unambiguous symptoms; procedures somewhat unclear; operators trained in correct actions.
0.0001	Well-practiced actions both in the plant and at the simulator; slowly changing transients with multiple chances for recovery by crew and others; clear indications and little chance of confusion; simple challenges, not multiple failures.

Parameter P_3 (Slips):

Screening Value	Conditions/Criteria
0.03	Multiple control board steps with nearby external actions.
0.03	Multiple control board steps; unpracticed actions.
0.01	Simple control board steps with nearby external actions.
0.006	Simple action but unpracticed.
0.006	Multiple steps but practiced.
0.002	Simple action, well-practiced.

The submittal (Table 3.3.3-8) lists the values of T_w and $T_{1/2}$ for in-control room actions, but provides very little discussion of the basis for the timing estimates. Timing studies using thermal hydraulic codes were performed to provide a basis for success criteria, and these apparently provided the basis for the estimates of T_w, the time available for recognition. In response to an NRC RAI, the licensee stated that values of $T_{1/2}$, time required by the operators, apparently were obtained from simulator observations and discussions with experienced training staff. Timing values for three critical actions were obtained by observing 10 operating crews in an ATWS simulator training event. Operating procedures and control board walkdowns were performed in advance of the simulator observations. (It is not clear from the licensee's response, but we assume that it was the HRA analysts, not the operators, who performed procedures and walkdowns prior to the observation. Obviously, the prior preparation by the operating crew would raise questions about the realism of the data.) All other time estimates were based on discussions with training staff. In response to the NRC RAI, the licensee indicated that these subjective estimates were questioned and revised during the interview process, and that no "correction factors" were judged to be necessary. (It is recognized that experienced operators tend to underestimate the time required for response, and some analysts/techniques, e.g., THERP, recommend multiplying such subjective time estimates by a factor of 2.)

2.3.4.2 Ex-Control-Room Response Actions. Table 3.3.3-9 in the submittal is a worksheet summarizing the technique, input parameters, and results for the SAIC approach used to calculate HEPs for 12 operator response actions taken outside of the control room. This table is the only documentation of the methodology in the submittal, and no external reference is provided. The methodology basically is a time reliability correlation (TRC) approach in which the HEP is calculated as a function of the ratio of time available to time required. The SAIC approach uses a family of correlations adjusted by several input parameters. The key parameters identified in the submittal table are the "Behavior Factor," which is set to 0.5 if the action is "rule-based" or 1.0 otherwise, and the "Burden Factor," which is set to 2 if a conflict exists, or 1 otherwise. Other parameters or calculated factors included in the formulation include a "Basic Error Factor," a "Model Uncertainty Factor," and a "Success Likelihood Index Factor."

The definitions and technical bases for these factors are not discussed in the submittal. The book "Human Reliability Analysis, A Systems Engineering Approach with Nuclear Power Plant Applications," by Dougherty and Fragola (Ref. 5) contains a detailed explanation of the SAIC TRCs. The formulation briefly summarized in the submittal table appears to correspond to the presentation in that text. For the BSEP analysis, it appears that the latter three factors are essentially "default" values that are part of the spreadsheet provided by Ed Dougherty of SAIC, who programmed the spreadsheet used to calculate HEPs. It does not appear that significant plant-specific evaluation was performed to support the quantification of the ex-control-room actions quantified by this approach. The primary plant-specific variables assessed were the time available and the estimated time required. The source for estimates of required time for ex-control-room actions is not discussed, but appears to be judgment by (or review of subjective estimates by) operations/training staff. The impression gained from the review of the submittal is that the use of this SAIC spreadsheet approach by BSEP staff was rather "mechanistic" in nature. That is, there did not appear to be a significant plant-specific investigation. To a certain extent, this is an inherent feature of the time reliability correlations. The HEP estimated is essentially driven by the time estimates, and other factors are either not considered or are assumed to be incorporated implicitly in the correlation data base. Therefore, the analyst does not obtain the benefit of an in-depth qualitative assessment of the plant-specific variables influencing human performance. The NRC RAI included questions to the licensee to clarify the licensee's understanding and use of the parameters in the SAIC model and the plant-specific analysis performed to support implementation of this methodology in the BSEP IPE.

The licensee's response to the RAI summarized the key parameters of the SAIC model and the basis for selection of values in the BSEP analysis (i.e., use of default values or plant-specific evaluation). In addition to the two time estimates discussed previously (time available and median response time), five parameters affect the HEP estimate. Two of these five, the behavior factor and the burden factor, were assessed on a plant- or case-specific basis. The behavior factor is one of the important factors determining which time reliability curve is selected for use in the model depending on whether the action is judged to be "rule-based" or "diagnosis-based". In the BSEP analysis, the factor was selected on the basis of procedure review and operator input. The burden factor, which differentiates cases with and without "conflict", also is a determinant in the selection of the time reliability correlation. Conflict is assumed to occur when the operators may be divided on what action should be taken or if the operators are reluctant to perform an action due to perceived consequences. Actions without conflict are given a higher chance of success. The burden factor was selected based on an evaluation of the specific action. Default values for "average well-trained crew" were assumed for the other three factors - "basic error factor", "model uncertainty factor", and "Success Likelihood Index. The first two of these factors are essentially uncertainty factors related to statistical variation in operator response and to modeling uncertainties. Selection of the default value is effectively not treating these uncertainties. The Success Likelihood Index (SLI) is a composite factor which permits the analyst to assess individual performance shaping factors, such as training or environment, and then apply the SLI to adjust the time reliability curve. Use of the default value implies that no outside influences were sufficiently important to alter the basic TRC curves. The licensee does not discuss the basis for this decision. However, it should be noted that HRAs often have used published time reliability curves without adjusting them for plant-specific evaluations. Indeed, most published time reliability curves do not offer as direct a means for parametrically modifying the curves. In summary, the licensee's response helps to clarify the assumptions made in applying the SAIC method in the BSEP IPE, and to clarify further the degree of plant-specific assessment performed. Based on our documentation-only review of the submittal and the licensee's response to the NRC RAI, it appears that the BSEP analysis did involve some plant-specific assessment, but in general employed generic data. Further, the "assessment" usually was limited to judgment by the IPE team members and/or operators. This does not imply that the numerical results of the HRA are any less (or more) realistic than other results of other HRAs. However, opportunities for learning about plant-specific influences obviously are proportionate to the degree of rigorous and in-depth plant specific analysis.

2.3.4.3 Operator Errors Modeled in the Event Trees. Eight actions were modeled in the event trees. All but one of these eight are actions involved in ATWS sequences. The submittal discussion of the methodology for quantifying HEPs these actions (paragraph 3.3.3.2.2, page 3.3.27) is somewhat unclear. The submittal states that quantification of HEPs for these actions was based on information taken from the results of a survey of the values for similar actions shown in previously published PRAs, and from the EPRI methodology discussed above. The survey specifically references the Shoreham PRA and the Limerick PRA as the PRAs included in this "survey." The submittal also states that, "Where possible, the HEP estimates also drew from BSEP operating experience." However, Table 3.3-11 of the submittal, which lists the 8 operator errors modeled in event trees, also lists the source of the HEP estimate, and the EPRI document is listed as the source of seven of the eight. The other HEP is identified as a "screening" value, which is based on the Peach Bottom and Shoreham PRAs. There is no further discussion of any data estimates based on BSEP data. Thus it appears that these errors were quantified primarily using the EPRI methodology and therefore with consideration of the timing and performance shaping factors as discussed above in Section 2.3.4.1 for the control-room actions modeled in the fault trees.

2.3.4.4 Recovery Actions. A number of different methods/sources were used for estimation of recovery action HEPs. Two of the eleven actions were quantified using the RMIEP model for recovery actions (Ref. 3). The RMIEP method employs time reliability correlations developed from simulator data obtained at the LaSalle training facility. Operator actions are grouped according to their similarity, primarily the judged difficulty of diagnosis. A plant-specific analysis should be used to determine the appropriate group(s) of LaSalle data (if any) to use. The time required to perform the action is estimated, in the case of BSEP on the basis of expert judgment. Time available for diagnosis is obtained by subtracting the required action time from the total time available. The HEP is then obtained from the appropriate time reliability correlation. There is no discussion in the submittal of the details of any plant-specific evaluation. The submittal does note that RMIEP groups 1 through 4 and group 12 were used to quantify the BSEP actions, and provides a listing of the HEPs as a function of time. The licensee's response to an NRC RAI indicated that consideration was given by the licensee to the applicability of the RMIEP data to these specific actions. The licensee contends that the relatively long time available for diagnosis and action (at least one hour) suggests that the probability for error in execution is relatively unimportant, and the probability of failure in diagnosis dominates the HEP. It is not clear to us that this is the case. In fact, it is more typical that the probability of failure to correctly diagnose the situation decreases with time, and execution errors may dominate the overall failure probability. However, the licensee also notes that for times greater than 60 minutes, the estimated HEP for diagnosis failure is the same (1E-03) for all RMIEP groups, and the choice of group does not greatly influence the final HEP value selected. From our review of RMIEP, it does not appear that literally all HEP values are the same beyond 60 minutes, but it is true that most of the RMIEP data is valid for periods less than 60 minutes and that in general, RMIEP data suggest an asymptote or "cutoff" at some lower value such as 1.0E-03.

Three local (ex-control-room) manual actions to recover motor operated valves (MOVs), were quantified using data from studies of MOV failures. A value of 0.5 was assigned to one action. The reference cited for this value is NUREG/CR-1368, which was a review of MOV failures to operate, which indicated that approximately 50% could have been opened manually. A value of 0.25 was assigned to a second MOV manual action on the basis of an EPRI study of MOV dependent failures (EPRI NP-3967) which indicates that approximately 25% of the events could be recovered by manual operation. A third local MOV recovery action was assigned an HEP of 0.1 because the failure of the valve to operate is a command fault, which is assumed to have a much higher probability of successful manual operation than for the cases that were not command faults.

One MOV recovery action, which could be accomplished from the control room, was evaluated using the EPRI methodology discussed previously. This appears to be the only case in which the quantification technique employed an analysis of plant-specific performance shaping factors.

Three operator actions related to recovery of offsite power - within 1/2 hour, within 2 hours, and within 7 hours - were quantified using values reported in NSAC-144, which was a study performed by SAROS, Inc. for the Brunswick PRA.

Two recovery actions, failure to recover Nuclear Service Water short-term and long-term, were quantified using "screening" values, the source of which was not identified in the submittal. The HEP values were 0.5 and 0.0001, respectively, for recovery short-term and recovery long-term. The definition of short-term and long-term is not provided in the submittal. The licensee's response to an NRC RAI indicates that the screening values were based on qualitative assessment and judgment by the PRA and HRA analysts. Time available was a major factor considered. The short-term action requires restoration of the service water system "within minutes" in order to assure continued core cooling. Without specific details, it is not possible for us to comment on the "realism" of this value. The value of 0.5 is typical of screening values used for post-initiator response-type actions, which are part of the "expected" response to an event, e.g., per the EOPs. The time available for the long-term action is several hours. The value of 0.0001 is more typical of a nominal value that would be supported by more detailed analysis.

The information discussed above on the various methods and data sources for quantifying recovery actions is contained primarily in table 3.3.3-12 in the submittal. There is little discussion of the basis for selection of different techniques/sources. The impression is that a reasonable attempt was made to obtain data or use techniques that were appropriate for the specific action under consideration, but that there was little plant-specific evaluation of performance shaping factors, including timing, etc. which would provide plant-specific insights. The emphasis in the submittal is on obtaining a reasonable numerical value, not

Overall, we consider this treatment of dependencies to be limited. Typically, the response to an accident event is considered to involve a highly dynamic human-system behavior in which each human action is highly dependent on the context of the situation - equipment response, environment, actions of other team members, and previous individual/crew actions. HRA analysts usually have identified dependencies among multiple human actions in cutsets, and typically these are treated quantitatively, e.g., by increasing the probability of dependent actions. It is positive that the licensee was able to respond that dependencies had at least been considered qualitatively. However, there does not appear to have been a significant quantitative assessment of the impact of dependent actions. While the screening process may have been sufficient to assure that important cutsets were not eliminated prematurely, the limited treatment of dependencies may have led to lower values of CDF than is appropriate. It is not possible for us to assess the impact without more detailed investigation which is beyond the scope of this review.

2.3.4.6 Treatment of Operator Actions in the Flooding Analysis. The submittal discussions of the analysis of internal flooding (Section 3.3.8) indicates that the licensee took credit for operator action to isolate the source of flooding. Ten flooding sequences were identified as potentially significant. An event tree was constructed for each of these sequences which consisted generally of 1) the flooding initiator, 2) operator or automatic action to isolate the flood source, and 3) alternate methods of coolant injection and decay heat removal. In general, it appears that operator action to isolate the flooding source was assigned a value of 0.01 to 0.001, based primarily on the time available for operator action. The submittal (Table 3.3.8.6) states that failure of isolation of internal flooding within times equal to or less than 2.4 minutes was assigned a probability of 1.0, based on "judgment." No basis or methodology is cited for the estimated values of 0.01 to 0.001. Table 3.3.8.10 lists eleven operator actions to isolate flooding sources in various plant areas, along with the time available to overflow and the time until all ECCS equipment is flooded, or other key systems are disabled. Table 3.3.3-10, which lists HEP values, indicates that all HEPs (except one with HEP = 1.0) are 3.0E-03. The table indicates that all HEPs were quantified using the EPRI model. There are several cases in which the time available is less than or equal to 2.4 minutes. Other times range from 9 minutes to hours. No estimates of actual time required for operator action are provided, and no discussion of an approach to estimating required time is provided. Eight flooding sequences are listed with sequence CDF above 1.0E-08/year. The highest CDF value is 7.<-07/year for a service water line rupture with failure of long-term decay heat removal.

Section 3.4.2.2 of the submittal discusses the insights gained from a sensitivity study in which the total contribution from operator action in any given sequence was constrained to be no less than 0.1. Among the results was the fact that seven flooding sequences would be above the cutoff of 1.0E-08/yr, if the HEPs were increased. The submittal notes this as further evidence of the importance of credit for operator action in the flooding scenarios, but does not provide details regarding the specific actions or HEP quantification.

In response to an NRC RAI, the licensee provided substantial additional clarifying information on the consideration of human action in the analysis of internal flooding. First the licensee notes that each room of importance to the flooding study has level switches which will quickly detect flooding and provide an alarm in the control room. Manipulation of valves to isolate flooding can be accomplished from the control room. The task of identifying the appropriate valve(s) for a given alarm is not discussed. Presumably this is apparent from the room location and alarm, since the possibilities for isolation are limited (primarily service water system or condensate storage tank). The human error probabilities were calculated from the EPRI methodology used for other post-initiator response actions. Time available was estimated from calculations of the time for water level to reach the height of critical equipment. As indicated previously, the licensee indicated in response to a separate NRC RAI that estimated required time for operator response is based on judgment. While the details of the analyses are not available and are beyond the scope of this review, the overall process for consideration of human action in the analysis of internal flooding appears to have been reasonable and generally consistent with practice in other PRAs.

2.3.5 GSI/USI and CPI Recommendations.

The licensee's consideration of generic safety issues (GSIs) and unresolved safety issues (USIs) and consideration of containment performance improvements (CPI) recommendations are the subject of the front-end review, and back-end review, respectively. The licensee addressed some aspects of decay heat removal, and indicates that unresolved safety issue (USI) A-17 is addressed through the analysis of internal flooding. Operator actions associated with DHR are identified as important actions and are discussed in Section 2.4.2 below. The back-end reviewer noted that the licensee concluded that with planned modifications associated with a hardened wetwell vent, no further changes were required to resolve the Mark-I containment issues. These modifications were not credited in the IPE.

2.4 Vulnerabilities, Insights and Enhancements

2.4.1 Vulnerabilities.

The submittal did not provide a specific definition of a vulnerability. In response to an NRC RAI the licensee stated that a project team was formed to assess IPE results and determine actions to be taken (if any) to address findings. The team used "both qualitative and quantitative criteria in successive levels of screening to determine whether enhancements appeared necessary and, if so, determine the most effective enhancement." The quantitative criteria used were the NUMARC 91-04 guidelines, applied first to accident sequences and then to accident classes. The qualitative criteria applied in parallel included cost-effectiveness and impact on plant operations of potential fixes, as viewed by corporate and plant engineers and operations staff. No vulnerabilities were cited. Enhancements are discussed in Section 2.4.3 below. In general, the process used to identify potential vulnerabilities/enhancements is consistent with approaches used in other IPEs.

2.4.2 IPE Insights Related to Human Performance.

The total core damage frequency (CDF) estimate for internal events and internal flooding for BSEP is 2.7E-05 per yr. The sequences with significant contribution to CDF (above the cutoff of 1E-08/yr) are listed in Table 2-4. The primary contributors are station blackout sequences, which contribute 66%, and transients involving loss of decay heat removal, which contribute 30%.

Table 2-4
CDF Contribution by Sequence

Sequence	CDF (/yr)	% Total
Station Blackout	1.8E-05	66
Transient With Loss of Decay Heat Removal	8.3E-06	30
Anticipated Transient Without Scram	7.0E-07	3
Transient With Loss of High Pressure Injection	3.1E-07	1
LOCA	1.6E-07	<1
Interfacing System LOCA	3.8E-08	<<1

2.4.2.1 Important Operator Actions. The submittal does not report importance calculations or a listing or discussion of the most important operator actions. Table 2-5 lists operator actions identified as important from our review of the submittal and/or by the NRC front-end reviewer. It also provides some comparison of Brunswick HEP values to values in other PRAs. (The absence of a corresponding action in other PRAs does not necessarily mean that the other PRAs did not address this operator action; we simply were not able to identify with certainty an appropriate action modeled at the same level of detail with the conditions similar to the BSEP action.)

Station Blackout. Operator actions important for the station blackout are recovery of offsite power and use of the cross-tie to the opposite unit to restore power to emergency buses. The Brunswick HEP values for failure to recover offsite power are typical of values used in other PRAs for comparable time periods. However, for BSEP, due to the relatively short battery life at least one emergency bus must be recovered within 2 hours. The higher HEP for this short time period is a significant factor leading to the high contribution from station blackout sequences.

Decay Heat Removal. Operator actions also play an important role in the contribution from sequences involving loss of decay heat removal. Operator response or recovery actions are involved in each of three decay heat removal options: 1) RHR in the suppression pool cooling mode; 2) reestablishing the condenser as a heat sink while using the condensate pumps to supply cooling water; and, 3) venting containment.

"Operator Fails To Correctly Initiate Suppression Pool Cooling Loop," (RHR-XHE-FO-SPC; HEP = 2.30E-04) is a control room response action quantified using the EPRI method and modeled as a basic event in a fault tree. It represents several manual alignment and switching actions that are required to initiate the suppression pool cooling mode of decay heat removal.

Use of the condensate system for DHR involves operator action to: 1) reopen the MSIVs, if they are closed; 2) re-establish circulating water cooling flow to the condenser; and 3) restart the condensate pumps. It appears that these operator actions were modeled as two groups. Actions 2 and 3 appear to be associated with "Operator Fails to Re-establish Vacuum," (CND-XHE-FO-OFF; HEP=7.00E-03; reopening MSIVs is modeled as a separate action in a number of sequences (e.g., "Failure To Reopen MSIVs for TM," PM3; HEP=5.00E-02).

The action "Operator Fails To Vent," (CAC-XHE-FO-VENTG; HEP=3.07E-02) was identified during the IPE as an important contributor to CDF. There are multiple possible paths for containment venting. Use of each requires multiple operator actions, some of which are performed in the control room, and some out of the control room. The submittal notes that actions are proceduralized, but are fairly complex and not well practiced in training or in normal evolutions.

Operator action to vent containment was assessed as an area for potential improvement by the Severe Accident Issues Project Team. The proposed enhancements associated with planned installation of a hard vent capability were judged to substantially simplify the required operator actions, reducing them to two switch manipulations from the control room. In comparing the decay heat removal vulnerabilities and possible enhancements for Brunswick vs. those reported for the Cooper IPE, the submittal notes that in comparison containment venting is more important for Brunswick because less reliance is placed on recovery actions. The BSEP analysis takes credit only for restoration of MOVs, not for restoring pumps, batteries or failed diesel generators. Thus greater reliance is placed on operator action to vent containment.

2.4.2.2 <u>Sequences Below Cutoff Due to Credit for Recovery Actions</u>. In response to guidance in NUREG-1335 (paragraph 2.1.6.6) the submittal identified sequences which were below the cutoff of 1.0E-8/yr but would have been above the cutoff were it not for credit taken for human action. Sixteen sequences were identified by the licensee. (Specific sequences were not reported in the submittal). Seven of these sixteen were transients with failure of decay heat removal; seven were associated with flooding sequences; and two, with station blackout. The actions associated with decay heat removal were discussed above. The submittal notes that operator sensitivity to internal flooding sequences was heightened in 1992

Table 2-5

Comparison of Selected Brunswick Post-Initiator Operator Actions with Other PRAs.

OPERATOR ACTION	HEPs				
	BRUNSWICK	**PEACH BOTTOM NUREG-1150**	**IPE-1** [(2)]	**IPE-2** [(2)]	**IPE-3** [(2)]
Failure to recover off-site power. (0.5 - 1 HR) (2 - 2.5 HRS) (7 HRS)	4.74E-01 2.62E-01 5.40E-02	3.10E-01 9.60E-01 3.20E-02		1.92E-01 9.60E-02 2.30E-02	
Operator fails to use non ADS SRVs to depressurize. [(1)]	6.14E-03	1.00E-02	4.20E-02	6.90E-03	1.78E-02
Operator fails to correctly initiate suppression pool cooling (SPC).	2.30E-04	2.00E-01	7.00E-02		
Failure to inhibit ADS during ATWS. [(1)]	2.51E-02			5.10E-01	1.28E-03
Operator fails to vent. [(1)(3)]	3.07E-02	5.00E-01			
Operator fails to terminate vent before injection sources from SP lose NPSH. [(1)(4)]	1.07E-01				
Start standby liquid control system, given that ATWS and reactor vessel in not isolated. [(1)]	2.69E-03	2.00E-01	1.24E-02	2.80E-03	
Operator fails to re-establish vacuum. [(1)]	7.00E-03			8.00E-04	
Operator fails to control level switch. (CRD used for injection) [(1)]	6.10E-03			8.15E-04	
Operator fails to fully open F003. (CRD used for injection, flow restriction) [(1)]	3.47E-02			6.9E-04	

NOTES:
(1) These actions identified as important during the front-end review.
(2) BWR-4 IPE previously reviewed by NRC.
(3) Nature of plant-specific actions makes direct comparison difficult, potential for unidentified modifiers.
(4) This action is unique to Brunswick therefore there are no comparisons.

when the Annual Emergency Plan Exercise scenario was based on an internal flooding event from the PRA. The licensee's response to the NRC RAI associated with treatment of operator action in the flooding analysis indicated that heightened awareness of the importance of isolation of flooding brought about by training and exercises such as this, plus the presence of level detectors in the flooded rooms with associated control room alarms provide an adequate response to internal flooding, and no further enhancements are required. In both of the SBO sequences, the dominant human recovery action is the inter-unit cross-tie of the emergency bus after diesel generator failure. A proposed enhancement to this cross-tie capability will include modifications to permit remote operation, which is expected to increase operator reliability.

2.4.3 Enhancements

The licensee's process for screening for vulnerabilities and enhancements was summarized above in Section 2.4.1. The licensee's assessment included review of plant modifications already committed for installation which will address some of the issues identified in the IPE, in particular those discussed above related to station blackout sequences and loss-of-decay-heat-removal sequences:

- A hardened vent modification which will enhance the likelihood of venting and thereby reduce the potential for loss of decay heat removal.
- Development of new procedures and installation of equipment to cope with loss of DC power and station blackout.

Based on judgment and the use of Fussel-Vesely importance rankings from the IPE results, the licensee concluded that these enhancements already planned or in progress would reduce the CDF by at least 40% and would be sufficient to achieve target criteria based on NUMARC guidance. No additional enhancements resulting from the IPE were identified.

III. CONTRACTOR OBSERVATIONS AND CONCLUSIONS

The intent of the IPE is summarized in four specific objectives for the licensee identified in Generic Letter 88-20 and NUREG-1335:

(1) Develop an appreciation of severe accident behavior.

(2) Understand the most likely severe accident sequences that could occur at its plant.

(3) Gain a more quantitative understanding of the overall probability of core damage and radioactive material releases.

(4) If necessary, reduce the overall probability of core damage and radioactive material release by appropriate modifications to procedures and hardware that would prevent or mitigate severe accidents.

The intent of our document-only review of the licensee's HRA process is to determine whether the process supports the licensee's meeting these specific objectives of GL 88-20 as they relate to human performance issues. That is, whether the HRA process permits the licensee to:

(1) Develop an overall appreciation of human performance in severe accidents; how human actions can impact positively or negatively the course of severe accidents, and what factors influence human performance.

(2) Identify and understand the operator actions important to the most likely accident sequences and the impact of operator action in those sequences; understand how human actions affect or help determine which sequences are important.

(3) Gain a more quantitative understanding of the quantitative impact of human performance on the overall probability of core damage and radioactive material release.

(4) Identify potential vulnerabilities and enhancements, and if necessary/appropriate, implement reasonable human-performance-related enhancements.

The following observations from our review are pertinent to NRC's decision regarding whether the licensee met the intent of GL 88-20:

(1) The submittal and supporting documentation indicates that utility personnel were involved in the HRA, and that the walkdowns and documentation reviews constituted a viable process for confirming that the HRA portions of the IPE represent the as-built, as-operated plant.

(2) The licensee performed an independent review that provides some assurance that the HRA techniques have been correctly applied and that the documentation is accurate.

(3) Pre-initiator human errors, both restoration errors and miscalibration, were considered in the analysis. The approach for identification and selection of pre-initiator actions included review of procedures and discussion with plant personnel.

(4) The quantification process for both pre-initiator human actions involved some, but relatively limited consideration of plant-specific factors that could influence human error probability. Thus the analysis provides limited insight as to plant-specific influences on these potential contributors to plant risk. It is a positive finding that numerical results generally were consistent with other PRAs, and that pre-initiator human actions were identified as some of the important human actions. However, a more in-depth assessment of plant-specific procedures and practice related to routine operations, maintenance, test, calibration, surveillance, etc. would provide the licensee with an enhanced understanding of actual performance at BSEP.

(5) The analysis of post-initiator human actions was reasonably complete in scope. Both response-type and recovery-type actions were included. The process for identification and selection of actions to be quantified included review of procedures and discussion with plant personnel. The quantification process considered timing of operator actions and, to a limited degree, other plant-specific performance shaping factors that could influence human error probability. The understanding of plant-specific influences on human behavior is limited by the degree of in-depth plant-specific assessment.

(6) Dependency among multiple human actions was addressed by the licensee, but the discussions by the licensee suggest that the analysis was limited in depth and rigor. This issue may be simply a problem of lack of thorough discussion/documentation; or it could be a weakness of the licensee's methodology. Failure to account for dependencies could lead to overly optimistic estimates of the impact of human error in response to accident sequences. Individual HEPs were, in general, consistent with results in other PRAs; but it is not possible from this document-only review to assess the impact of dependency assumptions on the overall IPE results.

(7) The licensee employed a reasonable screening process for identifying vulnerabilities. Enhancements already planned were identified as providing significant reduction in the estimated core damage frequency. No additional human-performance-related enhancements resulting specifically from the IPE were identified by the licensee.

IV. DATA SUMMARY SHEETS

Important Operator Actions/Errors:

ACTION	HEP
Failure to recover off-site power	4.74E-01 (0.5-1.0 Hr) 2.62E-01 (2.0-2.5 Hr) 5.40E-02 (7 Hr)
Operator fails to use non ADS SRVs to depressurize	6.14E-03
Operator fails to correctly initiate suppression pool cooling	2.30E-04
Failure to inhibit ADS during ATWS	2.51E-02
Operator fails to vent	3.07E-02
Operator fails to terminate vent before injection sources from SP lose NPSH	1.07E-01
Start standby liquid control system, given ATWS and reactor vessel is not isolated	2.69E-03
Operator fails to re-establish vacuum	7.00E-03
Operator fails to control level switch (CRD used for injection)	6.10E-03
Operator fails to fully open F003. (CRD used for injection, flow restriction)	3.47E-02

Human-Performance Related Enhancements:

None related exclusively to IPE results; several enhancements already committed to are expected to reduce CDF contributions from Station Blackout and from Loss of Decay Heat Removal. These include:

- Installation of a system to facilitate remote operation of the inter-unit emergency bus cross-tie
- Installation of a hardened wetwell vent
- Development of new procedures for DC power recovery and station blackout.

The remote operation of the emergency bus cross-tie and the hardened vent modifications directly address human performance issues identified in the HRA.

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