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WATTS BAR NUCLEAR PLANT - UNIT 1

TECHNICAL EVALUATION REPORT OF THE (INITIAL) IPE SUBMITTAL HUMAN RELIABILITY ANALYSIS

FINAL REPORT

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1. INTRODUCTION

This technical evaluation report (TER) is a summary of the documentation-only review of the Human Reliability Analysis portion of the initial Watts Bar Nuclear Plant (WBN) Individual Plant Examination (IPE) submittal to the U. S. Nuclear Regulatory Commission (NRC) in September, 1992. 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 documentation-only review and of the WBN 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.

Subsequent to completion of our review, the license applicant submitted an updated IPE to NRC. We did not review the updated submittal in any detail. **Throughout this report, unless explicitly noted, the comments, findings, conclusions, and references are from the original submittal.**

1.1 HRA Review Approach

The document-only review approach for WBN 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 and the organization of the HRA documentation, including any obvious major omissions. Identify notable features of the plant, the overall IPE approach, or the HRA approach that deserve special attention. Identify and obtain references that may need to be reviewed or checked, and 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 PRAs (e.g., the SLIM methodology described in NUREG/CR-3518 and 4016) (Refs. 1 and 2). Finally, the detailed review involves an attempt to "track" the complete assessment of a few key operator actions through the

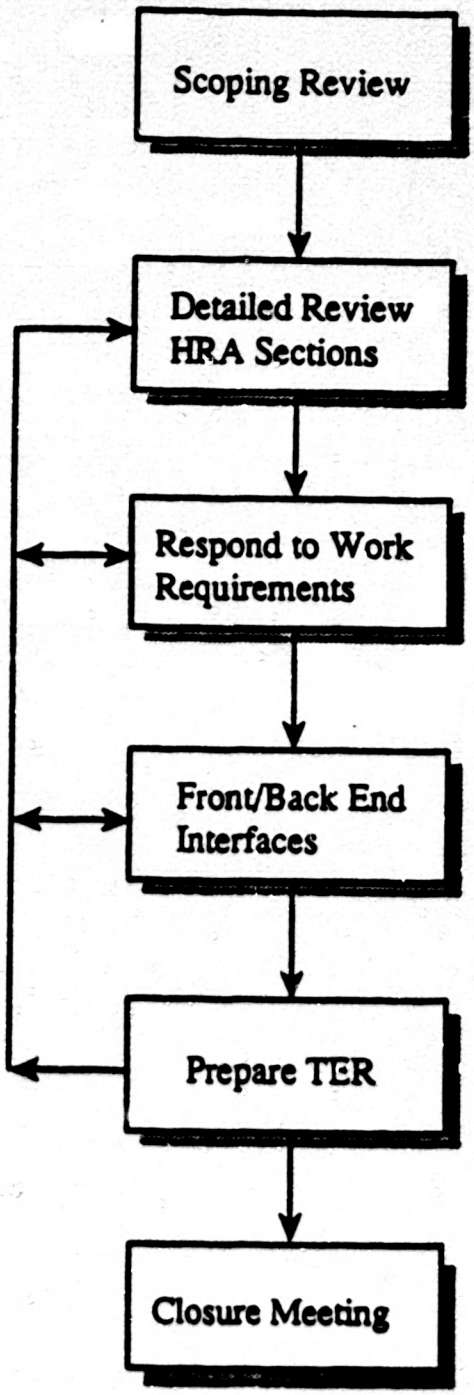

Figure 1 - Human Reliability Analysis Step 1 Review Approach

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 or verify details of quantitative analysis.

(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 obtaining insights regarding important results or potential areas of improvement. Any questions that require additional input from the license applicant 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 license applicant (if any).

1.2 Watts Bar Nuclear Plant IPE HRA Approach

The WBN HRA addressed both pre-initiator and post-initiator human errors, with emphasis on the latter. Quantification of post-initiator errors was performed using an adaptation of the Success Likelihood Index (SLIM) methodology for post-initiator actions, and using THERP for pre-initiator actions. The PLG method calculates a "Failure Likelihood Index (FLI)", rather than a Success Likelihood Index, but follows the SLIM methodology very closely. The methodology is essentially a systematic process for eliciting and quantifying expert opinion on likelihood of success (failure) in performing human actions. Evaluators (licensed operators and others directly knowledgeable of plant and operator responses) were provided with information on the accident sequence, operator actions being evaluated, and the major factors influencing likelihood of failure. The evaluators provided ratings of the impact of those factors on the likelihood of success, and these likelihood estimates were converted to human error probabilities via a scaling process using scale anchors or "calibration tasks". The

calibration tasks were selected to be similar to the WBN tasks. The submittal discussion identifies the performance shaping factors considered, the structured operator survey format, the process for rating PSFs, and the process for calculation of HEPs.

For pre-initiator operator actions, screening values were obtained from THERP (Ref.3) and used for initial values to assess operator actions that were judged to potentially disable systems. Nominal values were then estimated, also using THERP, for one action that had a potentially significant impact and warranted more detailed analysis.

2. CONTRACTOR REVIEW FINDINGS

The subsections below address explicitly 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, with regard to the specific work requirement and the overall intent of Generic Letter 88-20.

2.1 General Review of the HRA Approach.

2.1.1 Completeness of the Submittal With Respect to the Type of Information and Level of Detail Requested in NUREG-1335.

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 general methodology for accident sequence selection, accident sequence development, system modeling, HRA, and accident sequence quantification is described in Section 2 of the submittal. Section 2 also discussed plant damage states, and containment evaluation. The incorporation of human actions is reviewed in Sections 2.1.1 (3) and (4) below. The WBN HRA was performed using two methods. Because errors committed during pre-initiator human actions are dominated by failure to follow procedure, they were evaluated with THERP. The post-initiator human actions were evaluated with an adaptation of the SLIM method.

(2) Information Assembly. Section 2.4 of the submittal provides an overview of the information assembly and a review of the process used to obtain information. Similar PRAs and other documents were reviewed for possible PRA insights and other information. They are listed and discussed in Section 2.4.2 of the submittal. Plant documentation used to acquire information was listed in Table 2-8 of the submittal. Generic data was used for quantification of the plant model. The compilation and use of the generic component data is described in the appropriate portions of Section 3.3 of the submittal.

A more detailed description of information gathering process for the HRA analysis is described in Section 3.3.3 and Appendix B of the submittal. Plant documentation used to obtain information for HRA analysis was identified. Included were Plant Operating procedures, Emergency Operating Procedures (EOPs), and Surveillance and Maintenance procedures. A detailed description was prepared for each action to be analyzed by plant operators. Each description contains plant conditions and other constraints affecting the operator action.

This information was compiled on the Operator Response Form by HRA analysts and Licensed operators on the PRA team The completed Operator Response Forms were reviewed by the plant operations staff. Plant operators evaluate the PSFs using a structured method. Evaluation guidelines, as well as detailed information on the PSF breakdown and linkage to the survey rating system, were provided in the submittal. The SLIM-based evaluation process used plant operator

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. 2.1.6.6 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. 2.1.6.7 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.

input to evaluate PSFs, which were then converted to the failure likelihood index values. The survey process and information collection appear to be well structured.

(3) Accident Sequence Delineation. The process for development of the accident sequence models is described in Section 3.1 of the submittal. The initiator events were identified by review of similar Westinghouse PRAs and other sources identified, including NRC reports. The response to the plant upset was outlined in the Event Sequence Diagrams (ESDs). The ESDs were constructed by the PRA analysts, site engineers and operators familiar with the plant EOPs. The ESDs were formatted in a manner similar to the EOPs. The function of the ESDs was to ensure that the EOP actions were included in the accident sequence models along with the plant response. The ESDs were converted to the plant accident sequence models. The process and results of the process were detailed in the submittal.

(4) System Analysis. The System analysis is described in Section 3.2 of the submittal. An overview of the each system function and operation is provided. The top events or functions required by the Level 1 analysis are grouped into functional systems for analysis purposes. System notebooks were compiled to document the analysis. Included in each notebook are: the functional definition and success criteria of each top event, a description of system operation during normal and transient conditions, support system dependencies, references, assumptions used in constructing top event models, fault trees developed for the top event models, simplified system drawings, and quantification files. Operator actions incorporated into the system fault trees are pre-initiator events which affect system availability. Identification of operator actions is discussed in section 2.1.3 below. Documentation of the systems analysis appears to be sufficient to support a detailed evaluation, if one were necessary.

(5) Quantification Process. The overall description of the HRA effort in Sections 3.3.3 and Appendix B of the submittal provides a clear and appropriately detailed summary of the general methodology and approach to quantification of human actions within the IPE. The WBN HRA was performed using two methods. Because errors committed during pre-initiator human actions are dominated by failure to follow procedure, they were evaluated with THERP. Generic screening values were used for all but one pre-initiator action. The post-initiator human actions were evaluated with an adaptation of the SLIM method. Input pertinent to performance shaping factors (PSFs) was obtained from operator evaluation teams. Dependencies among multiple HIs in a sequence were identified as a part of the process of evaluation of PSFs by the operator teams. The structured process used to describe the operator actions and obtain operator input were summarized in appropriate detail in the submittal. PSFs used were described and justified. The calculation of HEPs based on operator input was outlined in the submittal.

(6) Front-End Results and Sequence Screening Process.

Front-end results and the sequence screening process are reported in Section 3.4 of the submittal. Key sequences with respect to CDF are discussed. Vulnerability screening is discussed in Section 2.4.1 of this TER.

The screening criteria used for reporting event frequencies and core damage frequency was based on guidance in NUREG-1335. The WBN PRA provides results in terms of systemic sequences, and reporting guidelines for systemic sequences were used. Criteria used for screening sequences to be reported were:

1. Any systemic sequences that contributes 1.0 E-07 per reactor-year or more to core damage frequency.

2. All systemic sequences within the upper 95% of total core damage frequency.

3. All systemic sequences within the upper 95% of total containment failure frequency.

4. Systemic sequences within that contribute to a containment bypass frequency in excess of 1.0 E-08 per reactor year.

5. Any other systemic sequence that the utility determines to be important to core damage frequency or to poor containment performance.

In addition, the core damage sensitivity to operator actions is discussed in Section 3.4. of the submittal and is further discussed in section 2.2.2 of this TER.

(7) Back-End Analysis - HRA Interfaces. The back-end analysis is described in Section 4 of the submittal. The interface between the front-end and back-end is a set of plant damage states containing the results of the front-end sequences. The back-end analysis addresses the physical progression of accident sequences from the onset of core damage through the release of radionuclides into the environment. The Containment Event Trees (CETs) consider the influence of the physical and chemical processes on changing the containment pressure and, when containment failure or bypass occur, on affecting the release of fission products from the containment. Operator actions are not explicitly included in the back-end analysis. The effects of the operator actions are represented in the plant damage states, which are the input into the containment event trees.

(8) Specific Safety Features and Potential Improvements. Section 6 of the submittal provides a discussion of specific safety features and potential improvements. Operator actions and plant hardware found to be beneficial to prevention of core damage and enhancement of containment performance were described. PRA screening criteria were used to identify potential enhancements. Improvements were identified. In addition, insights were derived based on sensitivities to various scenarios which did not meet the screening criteria discussed previously. These insights and recommendations are also provided in Section 6 of the submittal.

(9) IPE Utility Team and Internal Review. Section 5 of the submittal describes utility participation and the internal review process for the WBN IPE. The Risk Assessment Section (RAS) of the Corporate Engineering Department was responsible for the performance of the PRA on Watts Bar to meet the requirements for the IPE. The RAS team consisted of the project manager, leads for Level 1, Level 2, and data areas, electrical systems analysts, and systems

analysts. A licensed reactor operator was assigned to support the PRA effort. Table 5-1 of the submittal provides a matrix of individuals by organization participating in the IPE development or review by preparation phase. Organizations participating include Nuclear Engineering, Technical Support, Operations and RAS. Participants in the HRA analysis were not identified in this section. Outside contractors provided support for all facets of the IPE. Their participation in the IPE process is described in the submittal. PLG, Inc. supervised the application of their SLIM-based methodology and performed some of the HRA.

The internal review process described in the submittal appears to be extensive. Multiple engineers and operations personnel with expertise in Watts Bar design and operation were involved in the reviews. Table 5-1 of the submittal provides a matrix of individuals by organization participating in the IPE review by preparation phase. No individual was identified as the HRA reviewer or as having previous HRA experience. License applicant response to an NRC request for additional information indicated that the HRA did receive appropriate review by qualified operations personnel and experienced HRA analysts. (See Section 2.1.5 of this TER.)

2.1.2 Clarity of the Description of HRA Methodology.

Sections 3.3.3 and Appendix B of the submittal provide a reasonably clear and concise summary of the steps performed in the HRA portion of the IPE. Four types of human actions were considered: 1) Pre-Initiator Actions, e.g., restoration of equipment or instrument calibration performed as part of maintenance, test or surveillance activities prior to the initiation of an accident event; 2) Actions that cause initiating events; 3) Dynamic Operator Actions, accomplished during the plant response to an initiator; and, 4) Recovery Actions, which are human actions taken to recover failed equipment or provide an alternate means for accomplishing the intended function of failed equipment. (We refer to Dynamic Actions as "Response Actions"; both Response and Recovery Actions are referred to as "Post-Initiator Actions").

Human actions leading to initiating events were considered to be accounted for in the data base for initiating event frequencies, and are not treated directly in the HRA quantification. This is the practice in most PRAs.

Pre-initiator actions were quantified using THERP. Screening values taken from THERP were first applied to identify the most important pre-initiator actions. One pre-initiator action was found to be significant enough to warrant more detailed analysis. Screening values were retained in the IPE model for all of the others.

Post-initiator and recovery actions were evaluated using a PLG, Inc. adaptation of SLIM. The PLG methodology is similar to SLIM, which is thoroughly documented in NRC reports. The summary discussions in the submittal, in general, are clear, appropriately detailed, and reasonably complete. One area in which more information could have been provided was the database of operator actions and HEP values used for the calibration of WBN subjective estimates. The PLG database is contained in an unpublished PLG document, but presumably is available for review if necessary. In applying the SLIM-based methodology, plant conditions and other

considerations, such as preceding events, time constraints, required procedures, concurrent actions, competing factors and failure impact, were detailed for each operator action on an Operator Response Form. The evaluation process used groups of operators for evaluating specific PSFs and weighting factor which relates the relative influence of each PSF on the ability to perform the action. Detailed guidance for evaluation of the PSFs and weighting factors was provided in the submittal. The SLIM methodology has been modified by the PLG so that the operators scale the degree of difficulty. Conversion of these evaluated PSFs to the Failure Likelihood Index (FLI) value is accomplished by use of the weighting factors. The FLI value is converted to an error probability for each operator action. Reference actions are used to anchor, or "calibrate", the FLI for each action phase. Actions are sorted and grouped for conversion to error probability by PSF weighting factors. A separate quantification is done for each group of actions. Reference (calibration) actions for a particular group are chosen to match the actions in the group using similarity of PSF weights as the selection criterion.

2.1.3 Identification and Selection of Important Operator Actions.

The identification of pre-initiator actions is described in Section B.2 of the submittal. Qualitative screening of pre-initiator activities was performed to identify actions for analysis. Our review of the initial submittal noted that while calibration errors were cited as being considered, there was no discussion of specific calibration errors, and none of the HEPs listed were for calibration errors. A key concern is the potential for common cause instrument failure due to high dependence among individual calibration actions, for example, due to the fact that the same crew in a single shift may carry out all of the calibrations for a major function calibrations. The license applicant response to an NRC request for additional information stated that the potential for such common cause failures was addressed during the development of the individual system notebooks. The response indicates that calibration and surveillance procedures for each system were carefully reviewed to identify calibration errors that possibly could remain undetected and to include the potential for common cause errors among redundant trains. Section 1.5 of each system notebook documents the review. No potential common cause instrument failures due to calibration errors were identified.

Post-initiator operator actions were identified by the PRA analysts through the accident sequence delineation process detailed in Section 3.1 of the submittal. Event Sequence Diagrams (ESDs) were constructed by the analysts and plant operators to represent the integrated plant and operator response to plant upsets. The ESDs were the basis for the accident sequences developed for the plant model. The process for development of the accident sequences and event trees from the ESDs and the process for completing the qualitative description (Operator Response Form) are clearly summarized in the submittal. The Operator Response Form incorporates sequence-specific information about the operator action, which is used in the SLIM-based evaluation method. A summary of the Operator Response Form was included for each operator action.

Recovery actions were identified after the initial sequence quantification, and were quantified and incorporated in the model in the same manner as response actions. Recovery actions are discussed in Section 2.3.4 of this TER.

The process for identification and selection of human actions to be included in the model was appropriately summarized in the submittal and appears to be reasonably comprehensive. Comparison of operator actions quantified with those quantified in other accepted PRAs indicates that the assessment was reasonably comprehensive. We did not identify important operator actions that were overlooked.

2.1.4 Process To Confirm that the IPE Represents the As-Built, As-Operated Plant.

Section 2.4 of the submittal provides an overview of the information assembly and a review of the process used to obtain information. Approved plant documentation used to acquire information was listed in Table 2-8 of the submittal. The response to the any plant upset was outlined in detail in the ESDs. The qualitative descriptions for the human interactions were developed by a licensed operator who was a member of the PRA team. The completed Operator Response Forms were reviewed by operations staff for completeness. Also, the post-initiator operator actions were evaluated by teams of operators to assess the effects of PSFs included in the SLIM analysis. In addition, system notebooks were compiled to document the analysis. The system notebooks included: functional definition and success criteria for each top event, description of system operation during normal and transient conditions, support system dependencies, references, assumptions used in constructing top event models, fault trees developed for the top event models, simplified system drawings, and quantification files. The system notebooks were reviewed by site Engineering, Technical Support and Reactor Operations to ensure accuracy. The IPE was reviewed by the Watts Bar nuclear engineering, operations, and technical support personnel to ensure that the IPE represents the current design and operation of the plant as of December 1991. This process appears to be a reasonable and systematic approach to assuring that the IPE represents the as-built plant.

2.1.5 Independent Peer Review of the HRA.

The HRA was reviewed by a Watts Bar licensed Reactor Operator and a Sequoyah Senior Reactor Operator who served as consultants to the PRA team, by a staff member from PLG, Inc. with past experience in PRAs, including direct experience in performing HRA, by a staff member from the Corporate Engineering Risk Assessment and Safety Group with responsibility for the overall plant model, and by an independent consultant (Ian B. Wall) with extensive PRA experience. These reviews appear collectively to have provided a reasonable review process for the HRA.

2.2 Review of the Most Likely Sequences

2.2.1 Comparison to Other Accepted PRAs.

The human actions in the Surry NUREG-1150 study (Ref. 4) were compared to the WBN PRA human actions as a part of this review. The review shows that the appropriate important actions in the Surry analysis were included in the WBN analysis, and that additional operator actions were included in WBN analysis.

2.2.2 Accident Sequences Screened Out Because of Low Human Error Probabilities.

Section 3.4.3.2.3 of the submittal addresses the CDF sensitivity to operator dynamic actions. Dynamic operator actions error rates were increased to at least 0.1 and requantified. A brief discussion of new sequences that appeared above the 1.0 E-7 was provided. The key finding identified was that the new sequences identified usually involved two or more operator errors, which of course compounds the impact of changes in HEPs. This process of using a simplified sensitivity study to identify sequences that may have been screened out due to overly optimistic human error probabilities seems to be reasonable. The fact that additional sequences involved multiple operator actions emphasizes the importance of treating dependencies in those cases.

2.3 Review of the Quantitative Nature of the IPE Submittal.

2.3.1 Screening Values.

The identification and quantitative screening of pre-initiator activities was discussed in Section B.2 of the submittal. The screening criteria used were consistent with those used for pre-initiator actions in the Surry NUREG-1150 study (Ref. 4). Quantitative screening was performed for all identified pre-initiator actions. Screening values were taken from the THERP handbook. The removal of refueling canal drain plugs following maintenance, and the associated verification procedure were identified as warranting more detailed analysis. The THERP analysis was clarified by the license applicant in response to a request from NRC. An HEP of 0.003 was assumed for failure to remove the plugs, with recovery factors of 0.025 and 0.001 for two independent checks. The license applicant provided justification for selection of the THERP values used. While the THERP values were appropriately selected, the overall HEP of 7.5E-08 is very low in comparison to typical pre-initiator HEPs. All other pre-initiator errors were retained in the IPE model using the screening value.

No numerical screening was performed for post-initiator actions. All actions identified as important from the sequence and system analyses were quantified and included in the IPE model.

2.3.2 HEPs for Significant Human Actions

The human actions with the most significant impact on core damage frequency, in general, were the post-initiator response and recovery actions. Final (best-estimate) HEPs were developed for all post-initiator actions. Table 2-2 lists the most important operator actions (those contributing more than 1% to the total CDF). For completeness, we have listed the HEP values in both the original and the updated submittal. The operator actions are listed in their order of relative importance in the updated IPE model. The most important action in the updated model was not included in the original model. The overall core damage frequency estimated is reduced in the updated submittal. It appears that the primary reasons for the reduction relate to enhancements which rather dramatically reduce the contribution from sequences involving loss of component cooling water. While there were some general references to improvements in procedures, it appears from our cursory reading of the updated submittal that the updated values resulted

Table 2-2. Watts Bar Operator Actions Important to Core Damage Frequency
(Updated IPE Submittal, Table 3.4-7)

Operator Action	MEAN HEP (Original)	MEAN HEP (Revised)
Align ERCW to CCP 1A-A, 1B-B Unavailable	NEW ACTION	5.00E-01
Makeup to RWST after LOCA/Loss of Recirculation	4.41E-01	4.41E-01
Align HP Recirculation/Auto Switchover Successful	1.86E-03	5.33E-04
Start Turbine-Driven AFW Pump/Control or Start Signal Failure	8.25E-03	8.25E-03
Makeup to RWST/LOCA with Loss of Recirculation and Spray	7.21E-01	7.21E-01
Manually Start AFW - Reactor Trip with No Safety Injection (SI)	2.09E-03	2.09E-03
Identify and Isolate Ruptured Steam Generator	1.75E-03	1.75E-03
Cooldown and Depressurize RCS/SGTR - Isolation Failed	2.15E-02	2.15E-02
Refill CST During Non-Loca Events	3.63E-03	3.63E-03
Align and Start Alternate Cooling to CCP	1.61E-02	2.33E-01
Transfer to Hot Leg Recirculation - LOCA > 2" Diameter	1.78E-03	5.33E-04
Restore MFW/No AFW - No SI	3.88E-02	3.88E-02
Makeup to RWST after SGTR	2.46E-02	2.46E-02

primarily from different subjective estimates from new evaluator teams following the SLIM process, rather than from procedure enhancements.

2.3.3 Identification of Sources of Generic HRA Data and Performance Shaping Factors.

THERP was the source of generic human reliability data for the pre-initiator actions evaluated. As indicated previously, generic screening values were used for all but one of the pre-initiator actions, and the more detailed THERP analysis conducted for that action appears to be reasonable.

Post-initiator actions were evaluated using the PLG adaptation of SLIM. No generic data is used in this method. Operators are used as experts to rate and weight PSFs in relation to activities identified in the sequences. The Failure Likelihood Index (FLI) is then converted to HEPs using reference (calibration) values. The process for calibration of the FLI was detailed in the submittal and was discussed previously in this TER. The evaluation of the PSFs is well structured, and the justification for selection of PSFs is provided.

2.3.4 Clarity of Discussion of the Recovery Method and Reasonableness of Credit for Recovery Actions.

Operator recovery actions, (proceduralized and non-proceduralized) were identified by the PRA analysts after the preliminary sequence quantification. The most significant core damage sequences were examined for potential recovery actions. The recovery models included the development of a new event tree in which recovery actions were explicitly modeled. Recovery actions were discussed in section 3.1.3.2 of the submittal. The recovery actions were analyzed using the PLG SLIM-based method used for post-initiator activities. The process used for identifying and evaluating dominant sequence recovery actions appears well structured, and is clearly described. While the recovery actions are plant-specific and cannot be compared one-on-one to reference PRAs such as Surry, the approach to analysis appears to be reasonable, and the numerical values of estimated HEPs appear to be generally comparable to other PRAs. The methodology is well documented.

As indicated earlier in this TER, it is important to scrutinize carefully credit taken for operator actions that are not directed by procedures. The submittal indicated that non-procedural-guided actions were discussed with plant operations representatives to ensure that all actions assumed to be performed are compatible with the operating philosophy, and that qualitative descriptions (Operator Response Forms) were reviewed by plant operations staff. In response to an NRC request for additional information the license applicant provided a summary discussion of the eight recovery actions credited in the IPE that appeared to be non-proceduralized. The discussion summarized the importance of the recovery action (as determined from importance calculations using the IPE model), insights related to the SLIM-based analysis of the action, and other specific information regarding the expected operator response. The analysis and the credit taken for recovery actions appears to be reasonable.

2.4 The IPE Approach to Reducing the Probability of Core Damage or Fission Product Release.

2.4.1 Definition of and Screening for Vulnerabilities.

Vulnerability screening is described in Section 3.4.3 of the submittal. Vulnerability was defined to be when the mean CDF exceeds 5.0 E-4 per reactor-year. The basis for the CDF value is that other PWR PRAs using similar methods and data provide a range of 5.0 E-4 to 5.0 E-5. For Level 2 analysis vulnerability was defined as a large early release frequency of 5.0 E-5 per reactor-year. The basis for the value is a factor of ten benefit for containment integrity (value is a factor of ten below CDF value). If either value was exceeded, a vulnerability is identified only if a common function, system, operator action or other common element can be identified which contributes substantially to total frequency. No vulnerabilities were identified.

A second evaluation process, discussed in Section 6.3 of the submittal, involved examination of major contributors to either total CDF or the early release frequency to identify potential enhancements not associated with vulnerabilities. The set of screening criteria used for consideration of potential enhancements included the following thresholds:

Individual initiators, single component failures, or single operator actions	5.0 E-5 per reactor-year CDF
Single system train	1.0 E-4 per reactor-year CDF

Evaluation of the major contributors thus identified was directed at verifying the modeling assumptions and assuring that they were valid "outliers". The valid outliers were then evaluated for potential enhancements. These potential enhancements are identified in the submittal. The process described in the submittal appears to be a reasonable process for identifying vulnerabilities and potential enhancements (discussed in Section 2.4.2 below).

2.4.2 Human-Performance-Related Enhancements.

The screening process discussed in Section 2.4.1 above was applied, and the submittal lists three contributors to total core damage frequency that exceed the screening criteria for consideration of potential enhancements. The following potential enhancements to operator actions were identified for the three contributors, some of which were implemented by the time the updated IPE was submitted.

- Facilitate stopping RCPs on loss of CCS train A to minimize the potential for seal damage due to pump bearing failure.
- Clearer guidance on the desirability of cooling down the RCS prior to a seal LOCA developing to minimize seal damage.
- Additional training on loss of CCS initiator
- Provide connections for centrifugal charging pumps to ERCW system for lube oil cooling in the event of loss of CCS cooling (decrease of about 4% CDF).
- Revise procedures to reference fifth diesel generator following loss of offsite power and loss of both shutdown boards.

In addition, insights were derived based on sensitivities to various scenarios which did not meet the screening criteria discussed previously. These insights and recommendations are provided in Section 6.4 of the submittal. The process used to identify and evaluate potential improvements appear to provide enhancements which will improve operator reliability for those actions which were identified as important contributors to CDF.

2.5 Work Requirement 2.0 Complete data sheets.

Completed data sheets are included in Section 4 of this TER.

3. OVERALL EVALUATION AND CONCLUSIONS

This TER is based on our review of the original IPE submittal dated September, 1992. We performed only a cursory reading of the updated submittal. **Conclusions and findings are based on the original submittal.** On the basis of our review of the original submittal and the license applicant's responses to NRC requests for additional information, we conclude that the WBN HRA methodology provided the license applicant with a means to appropriately include in the IPE models the impact of human performance in severe accidents. The processes used to identify important actions, analyze factors influencing human performance, quantify human error, assess the impact of human error on system response (and therefore CDF and releases), and identify potential human-performance-related enhancements appear to be reasonable and consistent with practice in other accepted PRAs.

Utility personnel with operational experience were appropriately involved in the HRA process. Their involvement and review of HRA assumptions and analysis provided reasonable assurance that the HRA represents the as-built plant.

The independent review process conducted by the license applicant included specific review of the HRA by individuals with plant operations experience and with experience and qualifications in HRA. These reviews provided additional assurance of technical accuracy and appropriate implementation of the HRA methodologies employed, both of which are well recognized and thoroughly documented approaches.

The HRA addressed both pre-initiator and post-initiator actions. Post-initiator actions included dynamic "response" actions and "recovery" actions. Qualitative assessment of pre-initiator actions addressed both restoration errors and miscalibrations. Restoration errors were quantified using THERP, and the quantification process appeared to be reasonable. The license applicant determined that the potential for potential contribution from miscalibration did not warrant inclusion of those errors in the IPE model.

The SLIM-based approach used to evaluate post-initiator actions appears to be thorough, and was appropriately documented in the submittal. Additional detail on the data base used to anchor, or "calibrate,' the HEPs is desirable, but is beyond the level of detail implied in NUREG-1335.

Recovery actions were addressed using the same SLIM-based methodology as response actions. The analysis appears to have been reasonably thorough, and the credit taken for recovery actions appears to be reasonable.

The license applicant's process for identification of vulnerabilities appears to be consistent with guidance in NUREG-1335 and with processes used in other IPEs. No vulnerabilities were identified. Several human-performance related enhancements, in particular, related to improving operator response to loss of component cooling water, were identified. It appears that credit for implementation of these enhancements is one of the reasons for the significant reduction in estimated CDF in the revised IPE submittal.

4. IPE EVALUATION AND DATA SUMMARY SHEETS

Information Assembly

- List of plants, PRAs or other analysis known to have employed similar methodology.
  Seabrook, South Texas, Diablo Canyon

- Ex-Control Room actions treated? List.
  Ex-control Room actions were treated. Appendix B of submittal contains description of all operator actions.

Human Failure Data (Generic and Plant Specific)

- Analytical method used, e.g., Expert Judgment, THERP, SLIM-MAUD, HCR. TRC.
  - THERP for pre-initiator actions
  - Adaptation of SLIM based approach for post-initiator actions (failure likely index used)

- Were the following human errors considered:
  (1) Pre-initiator, e.g., maintenance error including testing, equipment calibration, and restoration. - Yes

  (2) Post-initiator procedural? - Yes

  (3) Post-initiator recovery
  - Control Room - Yes
  - Ex-Control Room - Yes

- Types of human errors considered, e.g. omission, commission
  - Errors of omission and commission.

- Source of human reliability data,

  Generic Data?
  - THERP Tables

  Simulator Data?
  - N/A

  Expert Judgment?
  - Operator input for adaptation of SLIM

- Most significant operator actions:

Operator Action	MEAN HEP (original)	MEAN HEP (Revised)
Align ERCW to CCP 1A-A, 1B-B Unavailable	NEW ACTION	5.00E-01
Makeup to RWST after LOCA/Loss of Recirculation	4.41E-01	4.41E-01
Align HP Recirculation/Auto Switchover Successful	1.86E-03	5.33E-04
Start Turbine-Driven AFW Pump/Control or Start Signal Failure	8.25E-03	8.25E-03
Makeup to RWST/LOCA with Loss of Recirculation and Spray	7.21E-01	7.21E-01
Manually Start AFW - Reactor Trip with No Safety Injection (SI)	2.09E-03	2.09E-03
Identify and Isolate Ruptured Steam Generator	1.75E-03	1.75E-03
Cooldown and Depressurize RCS/SGTR - Isolation Failed	2.15E-02	2.15E-02
Refill CST During Non-Loca Events	3.63E-03	3.63E-03
Align and Start Alternate Cooling to CCP	1.61E-02	2.33E-01
Transfer to Hot Leg Recirculation - LOCA > 2" Diameter	1.78E-03	5.33E-04
Restore MFW/No AFW - No SI	3.88E-02	3.88E-02
Makeup to RWST after SGTR	2.46E-02	2.46E-02

- Human Error contribution to core damage frequency (if known).
  - Quantification not provided.
- Vulnerabilities associated with human error.
  - None

PLANT IMPROVEMENTS AND UNIQUE SAFETY FEATURES

- Improvement insights stemming from HRA.
- Implemented human factor improvements or enhancements stemming from HRA.
  - None identified.

- Human factor improvements or enhancements under consideration.

  - Facilitate stopping RCPs on loss of CCS train A to minimize the potential for seal damage due to pump bearing failure.

  - Clearer guidance on the desirability of cooling down the RCS prior to a seal LOCA developing to minimize seal damage.

  - Additional training on loss of CCS initiator

  - Revise procedures to reference fifth diesel generator following loss of offsite power and loss of both shutdown boards.

  - Enhancements to operator training and procedures for responding to failures to support systems with emphasis on anticipation and coping with problems.

  - Provide additional guidance on relationship of CCS to ERCW and the desirability of eliminating CCS loads prior to CCS heatup.

  - Additional provisions for remote operation of S/G PORVs during loss of all AC power to depressurize S/Gs

  - Consider delaying spray operations to prevent rapid depletion of RWST and extends time for makeup

5. REFERENCES

1. Embrey, D.E.,"SLIM-MAUD: An Approach to Assessing Human Error Probabilities Using Structured Expert Judgment," NUREG/CR-3518, USNRC, March 1984.

2 Rose, E.A. et al., "Application of SLIM-MAUD: A Test of an Interactive Computer-Based Methods for Organizing Expert Assessment of Human Performance and Reliability," NUREG/CR-4016, USNRC, September 1985.

3 Swain and Guttman, "Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications," NUREG/CR-1278, August 1983.

4. USNRC, "Analysis of Core Damage Frequencies from Internal Events: Surry, Unit-1," NUREG/CR-4550/Vol. 3, Rev 1.

ATTACHMENT 5

SUMMARY OF THE WATTS BAR UNIT 1 INDIVIDUAL PLANT EXAMINATION (IPE)

SUBMITTAL ON INTERNAL EVENTS

Summary of the Watts Bar Unit 1 Individual Plant Examination (IPE) Submittal on Internal Events

The NRC staff completed its review of the internal events portion of the Watts Bar Unit 1 IPE submittal and associated information. The later includes an "Updated" IPE submittal and TVA's responses to staff generated questions and comments. The Watts Bar IPE submittal did not identify any severe accident vulnerabilities associated with either core damage or containment performance. However, the IPE did identify, and TVA implemented procedural enhancements which reduce core damage frequency by reducing the likelihood for reactor coolant pump seal LOCA.

Based on the review of the Watts Bar IPE submittal, the staff concludes that the licensee met the intent of Generic Letter 88-20.

The licensee's IPE results* are summarized below:

- Total core damage frequency (CDF) : 8.0E-5/Year
- Major initiating events and contribution to total CDF:

	Contribution
• Loss of Coolant Accidents	30.0%
• Loss of Offsite Power	23.3%
• Support System Faults	17.9%
• Internal Floods	11.3%
• Transients	7.7%
• Steam Generator Tube Rupture	5.0%
• ATWS	4.7%
• Interfacing System LOCAs	<1%

- Containment failure as a percentage of CDF:

	Contribution
• Intact	66%
• Late Failure	18%
• Bypass	5%
• Isolation Failure	5%
• Basemat Melt-through	4%
• Early Containment Failure	2%

Note: Intact containment includes failures after 48 hours.

- Major Contributors to Large, Early Release Frequency

	Contribution
• SGTR (with bypasses to the environment)	76%
• Containment failure due to direct impingement	15%
• α-Mode failure of vessel/containment	6%
• HPME/hydrogen burns at vessel breach	3%
• Hydrogen burns/DDT before and after vessel breach	<1%
• Interfacing system LOCAs	<1%

Note: Sum of Release Category Group I and SGTR bypasses from Group II

- Significant PRA findings:

  - No vulnerability was identified as per the screening criteria (Screening criteria per Generic Letter 88-20, Appendix 2)

  - Important plant hardware characteristics:

	Failure of	Contribution to CDF
o	Recirculation alignment	18%
o	EDG 1A-A	14%
o	EDG 1B-B	14%
o	CCWS train A	12%
o	RHR Pump train A	7%
o	RHR Pump train B	7%
o	480V shutdown board 1B1-B	7%
o	ERCW supply header 1B-B	5%
o	ERCW supply header 1A-A	5%
o	Reactor trip breakers	4%
o	Turbine driven AFW pump	4%
o	480V shutdown board 1A-A	4%
o	6.9KV shutdown board 1B-B	4%
o	6.9KV shutdown board 1B-B	4%

Note: The above percentages indicate the percentage of the sequences that involve the component under consideration and not the absolute percentage contribution of the specific function to CDF. Therefore, above percentages are not additive.

  - Important operator actions:

    - o Align ERCW to CCWS pump 1A-A, 1B-B unavailable
    - o Makeup RWST inventory following LOCA without sump recirculation
    - o Align for high pressure recirculation start turbine-driven AFW pump
    - o Makeup RWST following LOCA without recirculation and spray
    - o Manual start of AFW

- o Cooldown and depressurize RCS/SGTR
- o Refill CST during non-LOCA events

- Enhanced plant hardware, procedures, and operator actions:

  - o RCP pump trip on loss of CCWS train A to minimize the potential for RCP seal damage due to pump bearing failure - AOI-15, "Loss of Component Cooling Water"
  - o In the event of a total loss of CCWS, cooling down the RCS prior to a seal LOCA
  - o Hardware changes for Appendix R purposes and SBO for RCS cooldown
  - o In the event of a loss of offsite power followed by the failure of both shutdown boards on one unit, align the C-S diesel generator (i.e., the fifth diesel generator) to one of the shutdown buses not powered in the accident sequence due to the loss of a normally aligned diesel generator - AOI-35, "Loss of Offsite Power"
  - o Job Performance Measure (JPM) implementation to ensure that the plant operators are trained and familiar with various events for alternative measures
  - o New procedure for placing one train of the containment spray in standby prior to establishing high pressure recirculation
  - o New procedure to provide direction for cross-tying the 500KV offsite power to the 6.9KV shutdown boards of Unit 1 in the event of the loss of the primary 161KV offsite power supply

- Additional improvements under evaluation: None.