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SUBJECT: WATTS BAR NUCLEAR PLANT - COMPLETION OF INDIVIDUAL PLANT EXAMINATION REVIEW (TAC M74488)

See Reports

Dear Mr. Kingsley:

By letter dated September 1, 1992, TVA provided response to Generic Letter 88-20, "Individual Plant Examination (IPE) for Severe Accident Vulnerabilities." By letter dated December 27, 1993 TVA provided additional information requested by the staff in a November 10, 1993 technical meeting. Finally, by letter dated May 2, 1994, TVA provided updated information to the IPE.

We have completed our review of TVA's submittals regarding IPE, and conclude that TVA has met the intent of Generic Letter 88-20. Details are delineated in the enclosed staff evaluation. We note that although TVA recognizes the potential benefits of a probabilistic risk assessment (PRA) in future safety evaluations, TVA did not explicitly state that it plans to maintain its PRA as a "living" document. We believe that a "living" PRA could enhance plant safety and provide additional assurance that any potentially unrecognized vulnerabilities would be identified and evaluated during the life of the plant.

This completes our efforts on this review. The review was performed by Messrs. Jim Chung, John Lane and Erasmia Lois of the NRC staff, and contractors, Messrs. J. Jessen, J. Darby and R. Clark of Science and Engineering Associates, Inc.

Sincerely,

Original signed by

Peter S. Tam, Senior Project Manager
Project Directorate II-4
Division of Reactor Projects II-4
Office of Nuclear Reactor Regulation

Docket Nos. 50-390 and 50-391

Enclosure: Staff Evaluation

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50-390 TVA WATTS BAR

COMPLETION OF INDIVIDUAL PLANT
EXAMINATION REVIEW

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EXECUTIVE SUMMARY

The NRC staff completed its review of the Tennessee Valley Authority (TVA) Watts Bar Unit 1 IPE submittal for internal events and internal flood. The staff's review includes the original and updated IPE submittals, and TVA's response to the staff's request for additional information. The IPE is based on a Level 2 probabilistic risk assessment (PRA), with a containment analysis consistent with Generic Letter 88-20, Appendix 1.

Watts Bar Unit 1 is a Westinghouse four-loop, pressurized water reactor with an ice condenser containment. Similar plants are: Sequoyah, Catawba, and McGuire. TVA performed a probabilistic risk assessment (PRA) for the Watts Bar Unit 1 IPE using an integrated team of in-house engineers, contractor Pickard Lowe and Garrick (PLG), Inc, and consultants. TVA personnel familiar with the details of the design, controls, procedures, and systems of the plant provided PRA input and maintained involvement in the analysis and technical reviews of the PRA models. A subsequent IPE update, which supersedes the original submittal, was performed jointly by TVA and ERIN Engineering and Research, Inc.

The original submittal had reported a total Core Damage Frequency (CDF) of 3.3E-4 per reactor year (ry), 70% of which was due to reactor coolant pump (RCP) seal LOCA. The dominant sequence involved loss of the component cooling water system (CCWS) and subsequent operator failure to either trip the RCPs and/or provide RCP seal cooling in time to prevent RCP seal failure and associated LOCA.

TVA revised the IPE in order to incorporate plant design changes, procedure upgrades, and training enhancements. These items were completed in preparation of plant startup so that the updated IPE reflected the anticipated plant configuration at commercial operation. The most risk significant changes included:

- revision of the success criteria for the CCWS in order to credit one of two pumps as adequate for system success as opposed to two pumps (or one pump and operator action to isolate spent fuel pool cooling) which had been required in the original IPE,
- revision of the HRA reflecting updated procedures and training,
- revision of the "Loss of Component Cooling Water" procedure (AOI-15) and development of an associated Job Performance Measure for the operators,
- revision of the common cause analysis, especially for: CCWS, ERCWS and reactor trip breakers.

The revised and updated submittal reported a total mean CDF due to internal events of 8.0E-5/ry, which is approximately four times smaller than the CDF reported in the original submittal. The largest individual core damage sequence contributes less than 5% to the total CDF. The top 9 sequences account for 30% of the total CDF, and the top 104 sequences for 63%.

With respect to initiating events, LOCAs (which includes RCP seal failure as an initiator) contributes 30% to the total CDF, loss of offsite power contributes 23.3%, loss of support systems (which include loss of CCWS, ERCWS, and electrical power boards) contribute 17.9%, internal flood contributes 11.3%, transients (with successful reactor trips) contribute 7.7%, and steam generator tube rupture contributes 5%. Anticipated transient without scram (ATWS) contributes 4.7%, and interfacing system LOCAs contribute less than 1%. The most important sources of internal floods are associated with a rupture or flow diversion in the ERCWS trains.

It should be noted that, although the updated total CDF is reduced by a factor of four, the relative RCP seal LOCA percentage contribution to CDF did not change. RCP Seal LOCA in the revised IPE is associated with 76% of the total CDF. Of this 76%, approximately 22.5% is attributable to seal LOCA as an initiating event; 20% is attributable to loss of RCP seal cooling; and the remaining 33.5% involve RCP seal failures within the definition of the sequence but seal failure would not necessarily result directly in core damage.

TVA defined as "vulnerability" any core damage sequence that exceeds 1E-4/ry, or any mean large early release frequency that exceeds 5E-5/ry. Neither the original nor the updated submittal reported any vulnerabilities based on this definition. Therefore, no enhancements to specifically address vulnerabilities were identified. However, TVA searched for potential plant enhancements associated with individual initiators contributing to the CDF by more than 5E-05/ry, or a single system train failure contributing to the CDF by more than 1E-04/ry. The original submittal had identified three such contributors: loss of offsite power, total loss of CCWS as an initiating event, and failure of operator action to trip the RCPs in the event of a loss of CCWS train A. Several procedural and operator training enhancements were identified as a result of this evaluation including operator training that deals with loss of CCWS, and revision of the procedure AOI-15 "Loss of Component Cooling Water."

In accordance with the resolution of USI-45, the IPE specifically examined the decay heat removal (DHR) function for vulnerabilities. The original submittal documented and described contributors to DHR unavailability. No changes were performed during the IPE update on the DHR analysis. TVA concluded that there were no vulnerabilities associated with the DHR function.

Based on the review of the Watts Bar Unit 1 IPE submittals and associated documentation, the staff concludes that TVA met the intent of Generic Letter 88-20. This conclusion is based on the following findings: (1) the IPE is complete with respect to the information requested in Generic Letter 88-20 and the guidance document NUREG-1335; (2) the analytic approach is technically sound and capable of identifying plant-specific vulnerabilities, including those associated with internal flooding; (3) TVA employed a viable means to verify that the IPE models reflect the current plant design and start-up operation at time of submittal to the NRC; (4) the IPE had been peer reviewed; (5) TVA participated in the IPE process; (6) the IPE specifically evaluated the Watts Bar Unit 1 decay heat removal function for vulnerabilities; (7) the licensee responded appropriately to Containment

Performance Improvement (CPI) program recommendations. The staff notes, however, that, although TVA recognizes the potential benefits of a PRA and its potential use in future evaluations, the utility did not explicitly state that they plan to maintain their PRA as a "living" document. The staff believes that a "living" PRA could enhance plant safety and provide additional assurance that any potentially unrecognized vulnerabilities would be identified and evaluated during the life of the plant.

It should also be noted that the staff's review is a process review which, in general, is not intended to validate the accuracy of the IPE findings. Although certain aspects of the IPE were explored in more detail than others, the review primarily focused on the utility's ability to examine Watts Bar Unit 1 for severe accident vulnerabilities, and not specifically on the detailed findings (or quantification estimates) which stemmed from the examination.

I. BACKGROUND

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

All IPEs are to be reviewed to determine the extent to which each licensee's IPE process met the intent of Generic Letter 88-20. The NRC review of the licensee's IPE process may involve two steps: the first step (or Step 1 review) focuses on the completeness and the quality of the submittal; the second step (or Step 2 review) is a more detailed evaluation performed on selected IPE submittals. The decision to go to a Step 2 review is primarily based on the staff's need to determine the ability of the licensee's applied methodology to identify severe accident vulnerabilities and is determined on a case-by-case basis. As part of this review process, a Step 1 review was performed.

The NRC IPE review focuses only on the internal events portion including internal flooding. The licensee plans to provide a separate submittal on findings stemming from the IPE for external events (IPEEE). The IPEEE will be evaluated separately by the staff within the framework prescribed in Generic Letter 88-20, Supplement 4.

On September 1, 1992, Tennessee Valley Authority (TVA) submitted the Watts Bar Unit 1 IPE in response to Generic Letter 88-20 and associated supplements. On December 27, 1993, TVA submitted their response to staff's request for additional information (RAI), on May 2, 1994, submitted an updated IPE, and on September 29, 1994, provided a supplementary response to staff's RAI. As part of an ongoing effort to utilize and apply the IPE for Watts Bar Unit 1, on June 30, 1994 TVA submitted an assessment of potential cost effective plant enhancements and Severe Accident Mitigation Design Alternatives (SAMDAs). An evaluation of this June 30, 1994 submittal was not included in the staff's review of the IPE, as it is undergoing a separate review.

As part of its review, the NRC contracted with Science & Engineering Associates, Inc. (SEA), Scientech Inc./Energy Research Inc., and Concord Associates to review the front-end analysis, the back-end analysis, and the human reliability analysis, respectively. SEA's review is documented in the "Watts Bar Unit 1 Technical Evaluation Report On The IPE Front-End Submittal," Scientech's in "Step 1 Technical Evaluation Report Of The Watts Bar Unit 1 Individual Plant Examination Back-End Submittal," and Concord's in "Watts Bar Nuclear Plant Unit 1 Technical Evaluation Report Of The (Initial) IPE Submittal Human Reliability Analysis."

Section II of this report documents staff findings for each element of the IPE. Section III provides the staff's conclusions based on the consistency of the licensee's IPE process with respect to the Generic Letter 88-20 objectives and the Appendix summarizes key plant information from the IPE.

II. STAFF'S REVIEW

1. IPE Process

The staff examined the process used by TVA to perform the Watts Bar Unit 1 IPE. The IPE submittal documents and describes the techniques used to address each of the three major technical areas: the front-end systems analysis, back-end containment performance analysis, and the human reliability analysis (HRA). The methodology chosen for performing the analysis is consistent with the methods of examination identified in Generic Letter 88-20. TVA performed a Level 1 probabilistic risk assessment (PRA) for the front-end analysis, a Level 2 PRA for the back-end analysis, and used the "Techniques for Human Error Rate Prediction" (THERP) and the "Success Likelihood Index Method" (SLIM) in the HRA. TVA used contractor assistance in the performance of the IPE. Their principal contractor was Pickard, Lowe, and Garrick (PLG) Inc. They were also supported by ERIN Engineering, Inc.; Gabor, Kenton and Associates; and EQE Engineering, Inc. The contractors participated in the front-end, back-end and HRA portion of the IPE.

The submittal contained a summary description of the IPE program organization, the process employed, the participation of personnel, and the subsequent in-house peer review and their findings and conclusions. It appears that TVA used a viable process to ensure that the system and containment models represent the as-built and as-found plant. This process involved detailed document reviews, and walk-throughs, especially for the flood and containment analyses. The staff notes the considerable participation of the TVA personnel in virtually all aspects of the IPE through technology transfer, model development, reviews, data collection, and requantification of the models with plant-specific data. In addition to the TVA IPE team, other TVA and plant organizations were involved to ensure that the models accurately portrayed the plant. The original submittal notes that the IPE review process..."enhanced the plant personnel's overall understanding of PRA and its applications."

Within the scope of the original submittal of the IPE, TVA performed a comprehensive review and update of their IPE. The objective of the update was to incorporate plant design changes, procedure upgrades, and training enhancements that were made in preparation of plant startup so that the IPE reflects the "anticipated plant configuration at commercial operation." Another objective was to "refine" the analysis based on final design calculations in order to address both conservatisms and non-conservatisms of the previous IPE models.

From the changes performed during the IPE's update the most important are: (1) the Unit 2 systems for which the original IPE took credit under the assumption that Unit 2 is in full operation were re-evaluated and these systems are now considered as "transferred" to Unit 1; (2) the freeze date for plant data was expanded from December 1, 1991 to July 1, 1993 to include new

(last five years) industry experience and Watts Bar's startup and hot functional test phase experience; (3) excessive conservatism was either relaxed or eliminated, especially in the treatment of maintenance activities, system success criteria and common cause failures (CCFs); (4) human reliability data were revised to reflect plant, procedural, and operator training changes; (5) CCF data were changed on the basis of new system evaluations; and (6) success criteria were changed on the basis of pre-startup and hot functional test experience and the last five years industry experience that was not reflected in the original IPE. TVA identified the basis of the changes and documented them in a "Delta Report" (updated IPE, Appendix F). The staff finds the process used for performing and updating the IPE consistent with the Generic Letter 88-20.

2. Front-End Analysis and Decay Heat Removal (DHR) Evaluation

The staff examined the front-end analysis for completeness and consistency with acceptable PRA practices. The analysis capitalized on insights stemming from the Sequoyah NUREG-1150 study, a design similar to Watts Bar Unit 1.

The front-end IPE analysis used the large event tree/small fault tree methodology with linked event trees for sequence progression and quantification. The event tree linking was accomplished through a PC-based computer code, RISKMAN. The event sequences were modularized for the front-line and support systems. A support system event tree was used to evaluate support systems and their interactions with front-line systems. Key support system failures were identified through a failure mode and effects analysis (FMEA). Front-line system response to an initiator was determined with the aid of event sequence diagram (ESD).

Initiating events were identified using several approaches: comparison of categories from previous PRAs and industry studies; use of PLG database; review of Sequoyah NUREG/CR-4550 reactor trip summary; FMEA of plant support system events; and review of the Final Safety Analysis Report (FSAR). Forty-two initiating events were identified and categorized into four broad groups: (1) loss of reactor coolant inventory; (2) transients; (3) loss of support system; and (4) internal flooding. Each initiating event category identified leads to a plant trip. ATWS was addressed as part of the plant response scenarios developed for the four initiating event categories. Initiating events affecting more than one unit were not considered since the construction of the second unit has not yet been completed. The staff compared the list of initiators with lists from other reviewed PRAs, NUREG-2300, and NUREG/CR-4550. Staff also reviewed TVA's response to questions on initiating events concerning Control Air System, HVAC and very small LOCA.

To develop system and event sequence models, specific success criteria were defined for each major safety function with respect to each initiating event category. Detailed statements of system success criteria were included in the individual system analysis notebook. In the Update, some system success criteria were revised on the basis of the latest pre-startup and hot functional test experience and the last five years industry experience that was not reflected in the original IPE. In particular the success criteria for the component cooling water system (CCWS) in the original IPE were: two-of-

two train A pumps, or one pump and operator action to isolate the spent fuel pool heat exchanger or to shift cooling load to Unit 2 Train 2A. On the basis of new technical review of the CCWS design basis document (N3-70-4002, Rev. 3) this requirement was changed to one-of-two pumps. Also, on the basis of new calculations, conservatisms applied in the original IPE for room cooling requirements for AFW and CCWS pump rooms were removed in the updated IPE.

The plant sequence model included the responses of all front-line and support systems that are important to the prevention of core damage. The utilization of large linked event trees allowed the presentation of accident progression from the initiating event to the plant damage state(s). Event sequence models addressed dependency mechanisms such as common causes, intersystem, and intrasystem (intercomponent) dependencies. Shared systems between Units 1 and 2, particularly those systems credited in the IPE, were identified. Logic models for Unit 2 electric power systems supporting the shared systems were included. In the Update, the Unit 2 equipment and systems credited in the Unit 1 IPE are considered as transferred to Unit 1 since Unit 2 construction has been halted.

Special event trees were also developed. They are: the "containment interface trees" used to assign each accident sequence to an end state that reflects the plant conditions at the end of the Level 1 analysis; the "recovery event tree" used to model operator recovery actions applied at selected core damage sequences; and the "cross-reference of other special topics to event tree models" used for special topics regarding interfacing systems LOCAs, ATWS, and internal flooding. In general, the staff finds the Watts Bar Unit 1 event trees and special trees to be consistent with regard to initiating events, associated success criteria, and dependencies between top events.

The IPE explicitly identified physical and functional intersystem dependencies among plant systems, and provided dependency matrices addressing support to support and support to front-line systems dependencies. Major support systems analyzed include, but are not limited to, AC, DC, vital power, service and component cooling water, and HVAC systems. Control air system was not included in the support and intersystem dependency analysis. However, the portion of the air system classified as "safety-related" was evaluated in the containment response analysis. While plant heating, ventilating, and air conditioning (HVAC) systems were found important in the plant response during some event sequences, the submittal concluded that the loss of certain HVAC system was insignificant as an initiating event.

Because Watts Bar Unit 1 is not yet licensed and operating, the IPE's database is primarily generic, developed from the PLG proprietary database (PLG-500) which had been created from data of a large population of plants. Failure rate estimates contained in WASH-1400 (Ref. 14) and IEEE-500 were also used. Plant-specific features were taken into consideration for selecting appropriate generic distributions. Inductive methodology of Bayes' theorem was employed for "coherent" integration and updating the database. During the update, in order to capture the current plant status for the startup and hot functional test phase Watts Bar Unit 1 experience as well as recent (last five

years) industry experience, the data base was expanded from the original freeze date of December 1, 1991 to July 1, 1993.

Common cause failures (CCFs) were analyzed in two categories. The first category focused on system level failures and included sharing of common components, floods, and human errors during test and maintenance. The second category focused on the CCFs due to factors such as design errors, construction errors, procedural deficiencies, and unforeseen environmental variations. The quantification of CCFs was accomplished by using the multiple greek letter (MLG) method, consistent with NUREG/CR-4780. Although the MLG parameters and associated uncertainty distributions can be assessed using generic data, the lack of the plant-specific data precludes the direct identification and evaluation of possible CCF root-causes. In order to accomplish this for Watts Bar Unit 1, the analysts used statistical inferences for the frequency of the CCFs on the basis of the industry experience.

As mentioned before, changes were performed in the CCF analysis during the IPE's update. Specifically, in the original submittal the ERCW and CCWS pumps were treated as one group for CCF analysis; in the updated submittal they were treated separately. In addition, CCF factors for CCWS and ERCW were updated reflecting a more realistic plant configuration, and recent (last five years) industry operational experience that was not reflected in the original IPE. Also, on the basis of the last five years industry experience, the reactor trip breaker system mechanical failure CCF analysis was revised. The staff concluded that the IPE's treatment of CCF is consistent with NUREG/CR-2300 and NUREG/CR-4780.

The submittal describes the internal flooding analysis performed. A screening analysis of key safety equipment was performed to identify locations of potential flood sources and flow paths and, also, to identify mitigating features such as flood's drain, detection, and isolation. The analysts focused on modeling submergence-induced equipment failures. Spray-induced failure modes were considered to be localized and bounded by larger floods. The analysis included potential floods from feedwater line breaks outside the containment; floods that can cause an initiating event; floods that can cause CCFs of critical systems as for example failures of support systems; and floods that can occur during maintenance (while reactor at power), particularly large floods that may occur when valves and equipment are disassembled. Flood scenarios postulated in the IPE are based on screening and partitioning of industry events.

No flood scenarios leading directly to core damage were identified. However, ERCW-related large floods and subsequent system failures were found to have a critical impact on many other systems due to system functional dependencies. TVA reported a CDF of 9.1E-6/ry (11.3% of the total CDF) due to internal flooding. The most important sources of internal floods are associated with a rupture or flow diversion in ERCW trains. No changes were performed in the flooding analysis during the update. Based on the review of the description of the internal flood analysis in the original and updated submittal, the staff finds the flood analysis consistent with Generic Letter 88-20.

TVA had reported in the original submittal a total CDF of 3.3E-4/ry and had dealt extensively with the treatment of reactor coolant pump (RCP) seal degradation and failure due to loss of key support systems. In fact, RCP seal failures contributed approximately 70% to the total CDF. Seal failure due to total loss of CCWS accounted for the largest portion of this CDF, followed by loss of CCWS train A and failure of ERCW system. These results were based on: the success criteria that two-out-of-two pumps of CCWS train A were needed for the success; on the conservative CCF analysis for CCWS and ERCW; and the assumption that the operator will be inhibited to trip the RCPs on loss of CCWS.

The revised and updated submittal reported a total mean CDF due to internal events of 8.0E-5/ry, which is approximately four times smaller than the CDF reported in the original submittal. The largest individual core damage sequence contributes less than 5% to the total CDF. The top 9 sequences account for 30% of the total CDF, and the top 104 sequences for 63%.

With respect to initiating events, LOCAs (which includes RCP seal failure as an initiator) contributes 30% to the total CDF, loss of offsite power contributes 23.3%, loss of support systems (which include loss of CCWS, ERCWS, and electrical power boards) contribute 17.9%, internal flood contributes 11.3%, transients (with successful reactor trips) contribute 7.7%, and steam generator tube rupture contributes 5%. Anticipated transient without scram (ATWS) contributes 4.7%, and interfacing system LOCAs contribute less than 1%. The most important sources of internal floods are associated with a rupture or flow diversion in the ERCWS trains.

It should be noted that, although the updated total CDF is reduced by a factor of four, the relative RCP seal LOCA percentage contribution to CDF did not change. RCP Seal LOCA in the revised IPE is associated with 76% of the total CDF. Of this 76%, approximately 22.5% is attributable to LOCA as an initiating event; 20% is attributable to loss of RCP seal cooling; and the remaining 33.5% involve RCP seal failures within the definition of the sequence but seal failure would not necessarily result directly in core damage.

In accordance with the resolution of USI A-45, TVA examined the systems associated with DHR function to identify vulnerabilities as part of the IPE process. The evaluation was restricted to events initiated from power operation and hot standby, which is consistent with NUREG-1335. The original submittal contains a detailed discussion of the significance of the systems that directly perform the DHR function and includes system importance measures. Four classes of systems were considered: main feedwater (MFW), auxiliary feedwater (AFW), feed-and-bleed (F/B) cooling, and closed-loop residual heat removal (RHR).

In the original submittal, the most important MFW failures occur in sequences for which MFW is lost due to the same cause as the initiating event or due to the loss of support systems. Consequently, many of the main feedwater failures do not directly contribute to a sequence resulting in core damage. The contribution of MFW to CDF is 10%. Combination of failures of the AFW valves directing flow to the steam generators, with or without reactor trip,

contribute 1.4% to the CDF, and sequences involving failure of all three AFW pumps contribute 10%. Core damage sequences involving failure F/B are due to loss of support systems that preclude the operation of at least one PORV train; failure of F/B contributes 5% to the CDF. If one RHR train is operational, the contribution to CDF from all initiators involving the loss of closed-loop RHR is only 6%. Loss of both RHR trains contribute 56% to CDF; however, loss of both trains is attributed to loss of CCWS rather than the trains themselves.

TVA did not revise DHR evaluation in the updated submittal because no particular Watts Bar Unit 1 DHR system vulnerabilities had been identified. However, because of the changes in PORV requirements during the update and subsequent change in the top event success criteria, the total contribution to CDF by loss of F/B was reduced from 5% to 0.6%.

Based on the staff's review of the front-end analysis, the staff concludes that the employed analytical techniques are consistent with the NRC reviewed and accepted PRAs and capable of identifying potential core damage vulnerabilities. In addition, based on the staff's review of the process that was used to search for DHR vulnerabilities, and its review of plant-specific features, the staff finds the Watts Bar Unit 1 IPE/DHR evaluation to be consistent with the resolution of USI A-45. The staff, therefore, finds the IPE front-end analysis meets the intent of Generic Letter 88-20.

3. Back-End Analysis and Containment Performance Improvements Evaluation

The staff examined the Watts Bar Unit 1 IPE back-end (level 2) analysis for completeness and consistency with acceptable PSA practices. The containment analyses of Watts Bar Unit 1 were performed using the NUREG-1150 study of the Sequoyah ice condenser plant. After reviewing 111 top events of the Sequoyah accident progression event tree, the IPE analysts selected 30 top events to develop the containment event tree (CET) which they used to quantify the Watts Bar Level 2 analysis.

The analysts employed Revision 17.02 of the MAAP/PWR computer code to model the containment thermal response. Front- and back-end dependencies were addressed using 20 plant damage states with frequency greater than 1E-07. These plant damage states were chosen so that plant conditions, systems, and features that could have a significant impact on the potential course of an accident are represented. Included in the plant damage state specifications were important containment and system availability considerations, such as, the status of the containment (intact or failed) at the onset of core damage and the availability of engineered safety features, such as containment sprays, hydrogen control systems, and the ice condenser. The core damage plant damage state bins and associated containment system conditions resulted in 10 key plant damage states (9 plus V-sequence). These key plant damage states were subsequently analyzed by employing simplified but representative CETs. CET top events were developed using fault trees that represent the relationship of severe accident processes, systems operation, and operator actions. The CET end states were subsequently binned into four general release category groups:

- Group 1--Large Early Containment Failures and Large Bypasses - 2.4% conditional failure probability.

- Group 2--Small, Early Containment Failures and Small Bypasses - 10.1% conditional failure probability.

- Group 3--Late Releases and Long-Term Releases - 21.5% conditional failure probability.

- Group 4--Long-Term, Contained Releases - 66% conditional failure probability.

The radiological release estimates were calculated using MAAP. They included important parameters, such as, the time of release and the release fractions for 12 fission product groups, including noble gases, cesium iodide, and actinides.

A plant-specific containment strength analysis was performed to determine the probability of containment failure as a function of internal pressure and temperature for critical failure modes. The following modes were analyzed for containment overpressure failure, defined as an incipient leakage, for containment metal temperatures from room temperature to 800 degrees F:

- Cylinder hoop failure
- Dome membrane failure
- Equipment hatch buckling
- Containment anchor bolt failure
- Personnel hatch bulkhead flexure
- Baseslab failure
- Pipe penetration failure

Based on this analysis, a plant-specific probability failure distribution for a range of containment failure pressures was developed. The baseslab failure mode had the lowest median pressure capacity, followed by equipment hatch buckling, and dome membrane failure. Containment failure modes, in general, were found to be consistent with those identified in Table 2.2 of NUREG-1335.

The containment failure pressures for the various modes of failure were calculated as a function of containment temperature ranging up to 800 degrees F. As a result of this analysis, it was determined that only the three limiting modes of overpressure failure, dome membrane, equipment hatch buckling, and baseslab flexure, needed to be included in the CETs, although other non-pressure-related failure modes such as basemat meltthrough were also considered in the analysis. The temperature effect on equipment hatch buckling failure resulted in approximately a 20 percent reduction in median failure pressure as temperature ranges from 200 degrees F (117 psig) to 800 degrees F (98 psig). Personnel airlock/bulkhead failure was also analyzed to determine the effect of elastomer seal behavior at high temperatures. It was determined that the airlock and the bulkhead essentially maintained metal-to-metal contact, even after yielding of the construction material, SA 516 Grade 70 steel. Therefore, the performance of the elastomer seals was not a limiting factor in containment performance.

Containment isolation failure was analyzed using two top events in the CET: Top Event 1, "Containment not bypassed prior to core damage," and Top Event 10, "No containment failure prior to vessel breech." Top Event 1 covers pre-existing leaks, both large and small, which considers them to contribute towards an unisolated containment. The IPE treated containment isolation explicitly by developing a containment failure fault tree which included contributions from isolation failures. Purge lines isolate automatically upon signal and were, therefore, considered to be a small contributor to isolation failure. Approximately 2.6% of the total CDF included scenarios containing some form of small or large containment isolation failure.

The IPE used CETs to analyze all accident sequences (represented by plant damage states), that meet Generic Letter 88-20 screening criteria. The CETs integrated system/human response with phenomenological aspects, and allowed for recovery actions. For example, CET Top Event 3, "Core Damage Arrested Prior to Vessel Breach" estimated a split fraction of 0.23 for successful operator action to terminate core damage in medium LOCAs with available steam generator depressurization and low pressure recirculation.

The Watts Bar Unit 1 CET Top Events can be grouped into three categories. Top Events 1 through 12 represent the events before vessel breach. Top Events 13 through 22 and 29 represent events at or around vessel breach. Top Events 23 through 28 represent the events describing long-term vessel behavior.

Consistent with Generic Letter 88-20, Appendix 1, the analysts quantified the CETs by taking into account expected accident progression. The IPE addressed the most important severe accident phenomena normally associated with ice condenser containments including direct containment heating, seal table/containment liner failure, and hydrogen effects.

The analysts addressed uncertainties, by performing sensitivity studies recommended in the Electric Power Research Institute report, "Recommended Sensitivity Analysis for an Individual Plant Examination using MAAP 3.0B." In the Watts Bar Unit 1 IPE sensitivity was performed on: α mode vessel and containment failure, molten debris impingement on the containment wall, and hydrogen effects, such as the importance of igniters and air return fans.

In summary, TVA employed a process to understand and quantify severe accident progression. The process led to a determination of conditional containment failure probabilities and containment failure modes consistent with the intent of Generic Letter 88-20, Appendix 1. Sensitivity studies were performed to better understand the impact of phenomenological uncertainties and recovery actions. Dominant contributors to containment failure were found to be consistent with insights from other PRAs of similar design. For example, steam generator tube rupture, with bypass to the environment, was the dominant contributor to large, early release. The IPE characterized containment performance for each of the CET end-states by assessing containment loading.

Generic Letter 88-20, Supplement 3, contains CPI recommendations which focus on the vulnerability of containments, to severe accident challenges. For PWR ice condenser containments, such as Watts Bar Unit 1, the reference contains a recommendation that licensees evaluate the vulnerability to interruption of

power to the hydrogen igniters. A backup power supply meeting the requirements for the alternate AC option of the Station Blackout Rule was suggested as one method of ensuring uninterrupted operation of the hydrogen igniters.

The Watts Bar Unit 1 CET addressed the issue of containment failure from both early and late hydrogen burns. Using CET top events, the IPE addressed the need for backup power to igniters. The effect of backup power to igniters is important in the response to late burns inside containment. Early containment response without igniters will translate to the potential for late burns failing containment if a surreptitious spark were to ignite a globally flammable mixture late in a transient. Use of igniters early in a transient typically will not pose any early challenge to containment because the hydrogen concentrations are too low. Hydrogen phenomena were evaluated based on the MAAP burn criteria, which is consistent with the mechanistic model developed by DOE (Plys and Astleford, "Modifications for the Development of the MAAP-DOE Code--Vol. III," DOE/ID-10216, Vol. 3, November, 1988).

Sources of hydrogen for late burns include: in-vessel and ex-vessel oxidation of metallic zircaloy from the core, steel from the reactor vessel and internals, and steel reinforcing bars in the cavity and lower compartment. Only two KPDSs involved igniter unavailability and for these there exists the possibility of delayed ignition upon recovery of AC power. As a limiting worst-case condition to evaluate the efficacy of backup power to igniters, the analysts assumed that the recovery of AC power resulted in containment failure, if failure had not already occurred earlier. The combined frequency of these two KPDSs appear almost entirely in Release Category Group III, Late Failures, which typically involve small source terms. Thus, even under conditions where containment always fails when AC is recovered late in the accident, the contribution to Release Category III is relatively small, and the source term from this category is, itself, relatively small. By comparison, in a best-estimate calculation of this scenario, the calculated conditional containment failure probability for these conditions was found to be less than 0.001.

The staff's overall assessment of the back-end analysis is that the utility has made reasonable use of PRA techniques in performing the back-end analysis, and that the techniques employed are capable of identifying severe accident vulnerabilities. The staff also concludes that TVA's response to CPI Program recommendations, which included searching for vulnerabilities associated with containment performance during severe accidents, is reasonable and consistent with the intent of Generic Letter 88-20 and associated Supplement 3. Based on these findings the staff concludes that the Watts Bar Unit 1 IPE back-end process is consistent with the intent of Generic Letter 88-20.

4. Human Reliability Analysis

TVA identified and modeled two types of human events, pre-initiator human events associated with errors during "routine" activities (such as maintenance) leaving equipment disabled, and post-initiator human events associated with errors during operator response to an initiator. Post-initiator events were further distinguished into "dynamic" events associated

with operator response to an initiator according to plant emergency procedures (EOPs) (response-type events), and recovery events which are associated with operator tasks (according to EOPs or other procedures) needed to restore a completely or partially disabled system.

Pre-initiator human events were identified by reviewing equipment location, control room indications, and procedures (primarily surveillance) for determining conditions that would allow systems to be left in an undetected failed state. Screening guidelines were used to determine the more critical pre-initiator human events to be modeled. Based on these criteria, post-test or post-maintenance human events for which it was determined that adequate controls exist to assure their detection, were not modeled. Events that were modeled include realignment of components or flow path after testing, maintenance, or inspection, and removal of jumpers or other temporary alterations that restore equipment back to service.

The quantification of pre-initiator events was based on the THERP method (NUREG/CR-1278) which was used for developing generic human error probability (HEP) distributions and modifying them according to plant-specific performance shaping (PSFs). The IPE clearly describes how THERP was applied to derive the HEPs including the underlying hypotheses and assumptions used to modify generic HEPs according to plant-specific practices.

Of the pre-initiator events quantified, "failure to remove refueling cavity drain plugs following refueling" was found to be important to risk during an initial quantification where a screening HEP value was used. TVA performed a more detailed evaluation of the procedures associated with this task. The analysis indicated a failure rate of several orders of magnitude lower as more appropriate. Subsequently, this event was determined not to be a contributor to risk. The staff, however, expects that TVA will support long term operational practices (training, procedures) to ensure that the error probability is small.

Response-type events were identified by reviewing operating procedures to determine those operator tasks that will bring the plant to a safe shutdown. Identified tasks were evaluated qualitative on a scenario-by-scenario basis to ensure that they were appropriately identified and modeled. Recovery events were identified by reviewing the sequences contributing to the CDF after an initial quantification. The IPE identified some recovery events associated with tasks that are not part of the EOPs. It appears that during the evaluation of these post-initiator events, care was taken by the analysts to ensure that the modeling of these events were reflecting operator knowledge and training, as well as plant practices and procedures, appropriately.

The Watts Bar Unit 1 IPE employed a modified SLIM to quantify post-initiator events. The IPE discusses the important aspects of post-initiator event analysis, as, for example, the underlying assumptions for converting operator judgments into HEPs, the PSFs influencing (positively or negatively) the operator to perform a task, the machine-human and human-human dependencies, and the timing for each human action.

The HRA was revised during the IPE's update to reflect the changes performed in the IPE's plant model due to plant design and operations changes, as well as, changes in the IPE's underlying hypotheses. TVA established a set of guidelines to select the human events that needed to be revised. Furthermore, sessions were conducted with operators familiar with the original HRA, with current operations crews, and operators experienced in HRA results. As a result, TVA updated the HEPs for 15 post-initiator (response type or recovery type) human events.

Post-initiator quantitative results and insights derived from the analysis were discussed in a clear and concise manner. Two tasks that were identified in the original submittal as important: (1) Stop RCPs upon Loss of A CCWS or RCP Cooling Path" with a 19% contribution to the total CDF, and (2) Isolate CCWS Train A from Spent Fuel Pool Heat Exchanger with a 5% contribution to the total CDF, did not turn out to be important in the updated IPE. Their significance was reduced due to changes applied in the PRA model, primarily, in the systems' success criteria and in the credit taken for procedural and training improvements.

The most important human actions reported in the updated IPE included:

a. Align ERCW to CCP 1A-A, 1B-b unavailable
b. Makeup to RWST after LOCA/Loss of Recirculation
c. Align HP Recirculation/Auto Switchover Successful
d. Start Turbine Driven AFW Pump/Control or Start Signal Failure
e. Makeup to RWST/LOCA with Loss of Recirculation and Spray
f. Manually Start AFW - Reactor Trip with No Safety Injection

The IPE included importance and sensitivity analyses. One important finding of the updated IPE was that the action to "align and initiate alternate component cooling to the 1A-A charging pump (HCCSR2)" was significantly underestimated in the original IPE. The original mean HEP was 1.6E-2. During the update, the analyst initially calculated a value of 2.3E-1 because they identified that the operators were not sufficiently trained to perform this action during a loss of RCP cooling within the required time frame (8-10 minutes). In order to improve operator reliability for this action "significant procedure changes to AOI-15" (pg. 3.3-4) were performed. Based on the revised procedure, the HEP for this action was also revised in the update to a value of 8.9E-3. It should be noted that the revision of AOI-15 had been identified as a "potential improvement" in the original submittal. Furthermore, a new job performance measure was adopted for the operators ensuring they will be appropriately trained in the execution of the tasks directed by AOI-15.

The staff found the Watts Bar Unit 1 HEPs reasonable and compatible with HEPs used in other HRAs reviewed and accepted by the staff. The staff's review did not identify any significant problems or errors in the human reliability analysis. The overall assessment is that TVA has made reasonable use of the techniques in performing the human reliability analysis that is capable of identifying severe accident vulnerabilities. The staff, therefore, concludes that the Watts Bar Unit 1 IPE human reliability analysis meets the intent of Generic Letter 88-20.

5. Actions and Commitments from the IPE

TVA defined as "vulnerability" any core damage sequence that exceeds 1E-4/ry, or any mean large early release frequency that exceeds 5E-5/ry. Neither the original nor the updated submittal reported any vulnerabilities based on this definition. Therefore, no enhancements to specifically address vulnerabilities were identified. However, TVA searched for potential plant enhancements associated with individual initiators contributing to the CDF by more than 5E-05/ry, or a single system train failure contributing to the CDF by more than 1E-04/ry. The original submittal had identified three such contributors: loss of offsite power, total loss of CCWS as an initiating event, and failure of operator action to trip the RCPs in the event of a loss of CCWS train A. Several procedural and operator training enhancements were identified as a result of this evaluation including operator training that deals with loss of CCWS, and revision of the procedure AOI-15 "Loss of Component Cooling Water." Also, a design change which connects both centrifugal charging pumps lube oil cooling units to the ERCW system (which previously existed only for centrifugal charging pump A of Unit 1 at the time of the submittal), was identified as reducing the total CDF by 4%. TVA, however, did not commit to the implementation of these enhancements.

During the update, AOI-15 procedure was revised to facilitate stopping RCPs on loss of CCWS train A and to provide alternate cooling to RCP seals through the alignment of ERCW to centrifugal charging pump (CCP) train A. A new Job Performance Measure associated with ERCW alignment to CCP 1A-A was also implemented; additional procedural and training enhancements were identified and implemented; and a hardware change providing nitrogen bottles for use in providing pneumatics to the AFW flow control valves and steam generator PORVs was implemented enhancing operator ability to cooldown RCS during station blackout (SBO).

As noted under "Background," TVA also submitted an assessment of potential enhancements and SAMDAs using the updated IPE. As a result of this assessment, TVA determined that development of two new procedures was cost beneficial and stated that these changes would be incorporated in the operating procedures before initial criticality. The first procedure will direct operators to place one train of the containment spray in standby prior to establishing high pressure recirculation. This will provide additional time to the operator for aligning high pressure recirculation and for responding to hardware failures. The second procedure will 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. This enhancement will reduce the contribution from Station Blackout.

III. CONCLUSION

The staff finds the Watts Bar Unit 1 IPE for internal events, including internal flooding, consistent with the information requested in NUREG-1335. Based on the review of the original and the updated IPE submittals, the staff finds TVA's conclusion that no severe accident vulnerabilities exist at Watts Bar Unit 1 reasonable. The staff notes that:

(1) TVA personnel were considerably involved in the development and application of PRA techniques to the Watts Bar Unit 1 facility and that the associated walkdowns and documentation reviews constituted a viable process for confirming that the IPE represents the as-built, as-found plant.

(2) The front-end IPE analysis appears complete with the level of detail consistent with the information requested in NUREG-1335. In addition, the employed analytical techniques reflect commonly accepted practices and are capable of identifying potential core damage vulnerabilities.

(3) The back-end analysis addressed the most important severe accident phenomena normally associated with ice condenser containments, for instance, DCH, and hydrogen combustion. No obvious or significant problems or errors were identified.

(4) The HRA enabled an understanding of the contribution of human errors to CDF and containment failure probabilities.

(5) Based on the process used to search for DHR vulnerabilities and review of Watts Bar-specific features, the staff finds the IPE/DHR evaluation consistent with the intent of the USI A-45 (Decay Heat Removal Reliability) resolution.

(6) TVA's response to the CPI Program recommendations, which include searching for vulnerabilities associated with containment performance during severe accidents, is reasonable and consistent with the intent of Generic Letter 88-20, Supplement 3.

The staff also notes that TVA did not explicitly state that they plan to maintain their PRA "living." The staff notes that a "living" PRA could enhance plant safety and provide additional assurance that any potentially unrecognized vulnerabilities would be identified and evaluated during the life of the plant.

Based on the above findings, the staff concludes that TVA demonstrated an overall appreciation of severe accidents that could occur at Watts Bar Unit 1; has an understanding of the most likely severe accident sequences; has gained a quantitative understanding of core damage and fission product release; and has responded appropriately to safety improvement opportunities. The staff, therefore, finds the Watts Bar Unit 1 IPE process meets the intent of Generic Letter 88-20.

APPENDIX
WATTS BAR UNIT 1 DATA SUMMARY SHEET*
(IPE UPDATE, INTERNAL EVENTS)

- 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.

* The above information has been taken from the Watts Bar Unit 1 IPE Update and has not been validated by the NRC staff.

ATTACHMENT 2

WATTS BAR UNIT 1 INDIVIDUAL PLANT EXAMINATION

TECHNICAL EVALUATION REPORT

(FRONT-END)