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Gentlemen:

In the Matter of)	Docket Nos.	50-259
Tennessee Valley Authority)		50-260
			50-296

BROWNS FERRY NUCLEAR PLANT (BFN) - PROBABILISTIC RISK ASSESSMENT (PRA) - SUMMARY REPORT

- References:
1. G. E. Gears' summary of May 18, 1988 meeting with the Tennessee Valley Authority dated May 27, 1988
 2. S. D. Ebnetter's letter to S. A. White transmitting NRC staff comments on the January 1986 version of the BFN PRA, dated October 1, 1987

This letter transmits the information requested by reference 1, fulfilling the commitment to provide a summary document by August 30, 1988. As requested, the report includes:

*The rationale for concluding that the revised PRA will conservatively reflect the configuration of Unit 2 at the time of restart and

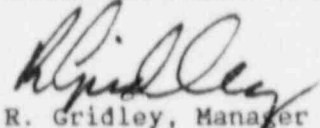
*A summary of the changes made between the January 1986 PRA reviewed by the staff in its October 1, 1987 letter (reference 2) and the September 1987 version.

The purpose of the summary report is to provide additional information to support a conclusion that BFN is not an outlier with respect to severe accident characteristics when compared to plants of similar type and vintage.

No new commitments are made with this transmittal. If you have any questions, please call D. L. Williams at (615) 632-7170.

Very truly yours,

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ENCLOSURE

Browns Ferry Nuclear Plant Probabilistic Risk Assessment
(BFN PRA)
Summary Report

As requested by the NRC staff (reference NRC letter dated May 27, 1988), this Summary Report provides:

- TVA's rationale for concluding that the latest BFN PRA will conservatively reflect the configuration of BFN Unit 2 at the time of restart and
- Summary of the changes made between the January 1986 version of the BFN PRA reviewed by the NRC staff in its October 1, 1987 letter and the September 1987 version.

The purpose of this report is to provide additional information to support a conclusion that BFN is not an outlier with respect to severe accident characteristics when compared to plants of similar type and vintage.

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1.0 EXECUTIVE SUMMARY AND CONCLUSIONS

The BFN PRA makes use of realistic modeling assumptions and plant specific data to the greatest extent practical. The level of detail of the analyses is consistent with that found in other PRAs performed by the industry. The total core damage frequency estimated for BFN Unit 2 is 4.7×10^{-4} /year, within the range typical for similar BWR plants. No sequence of events contributes greater than 5% to the total core damage frequency. No new generic safety issues have been identified by the analyses.

The September 1987 version of the BFN PRA conservatively reflects the configuration of BFN Unit 2 based on completion of a review of plant drawings and implementation of a review process to evaluate future changes to the plant design.

Changes made between the January 1986 PRA and the September 1987 version are primarily due to:

- Refinement of the models used,
- Analyses initially done as scoping completed in detail,
- Addition of more plant specific data, and
- Changes in assessment of operator actions.

2.0 INTRODUCTION

The BFN PRA is a full scope probabilistic risk assessment of the Browns Ferry Nuclear Plant which includes:

- Core damage accident sequence analyses,
- Containment (Mark I) response analyses, and
- Site specific consequence analyses.

Initiators considered include transients, loss of coolant accidents (LOCAs), and events such as earthquakes, fires, internal and external flooding, high wind, aircraft impact and turbine missiles. Systems and sequence analyses were performed in detail. Initiating events that could result in core damage occurrence were analyzed and systems needed to mitigate these events were modeled to estimate the core damage frequency.

Development of a PRA is an iterative process. During the continual review and improvement activities inherent to the PRA process, refinements to the system models, hardware data and initiating event frequencies are identified. Refinements also result from the ongoing evaluation of changes to the plant configuration and procedures as discussed in Section 3.0. The major changes incorporated into the BFN PRA September 1987 version are discussed in greater detail in Sections 4.0 and 5.0.

3.0 MAINTAINING PLANT CONFIGURATION IN THE PRA

A two-phase review program has been implemented to assure that the PRA will conservatively reflect the BFN plant configuration at restart. The first phase consisted of a drawing review to establish a design reference base from which the continuing change monitoring and review program of the second phase builds.

3.1 Confirmation of Design

A review of BFN drawings (Phase 1) was performed to determine if there were significant differences between the PRA model and the current plant configuration. Several sets of information were found which provided overlapping information. An evaluation was made to determine and select the most current set of information. An information hierarchy was established for the BFN Unit 2 drawings as follows:

1. As-Constructed and Verified
2. As-Constructed
3. As-Designed

A drawing "freeze" date of May 15, 1988 was selected. A list of drawings referenced in previous versions of the PRA was then prepared. The latest revisions of these drawings were obtained, filed, and declared "current" to May 15, 1988. Modifications to the plant and revisions to these drawings after this date are addressed in Phase 2, the change monitoring and review program.

The referenced drawing list was sorted by PRA system and distributed to the reliability engineers. Each engineer reviewed these drawings to identify and evaluate the significance of the differences between the PRA models and the current revision of the drawings. These differences were documented on a "Drawing Review Form." No changes were identified in Phase 1 which significantly influenced the results of the PRA.

3.2 Process for Evaluation of Future Design Changes

Phase 1 documented the accuracy of the PRA models up to the plant configuration freeze date of May 15, 1988. The change monitoring and review program (Phase 2) provides an overview of the changes in the plant configuration after the freeze date to determine any effect on the PRA models and conclusions. This program is not intended to continuously revise the PRA models to reflect the actual plant configuration. Instead, the program ensures that, between periodic updates, the conclusions reached as a result of using PRA analysis remain accurate.

Changes in the plant which involve physical modifications are documented by drawing revisions and through engineering change notices (ECNs). Revised drawings are distributed to and reviewed by reliability engineers for potential impact on BFN PRA conclusions. In a manner similar to the revised drawing review, the ECNs designated as "drawing complete" or "closed" are collected and processed through the change monitoring and review program.

The revised drawings and the ECNs designated as "drawing complete" or "closed" are combined into Change Evaluation Packages. Each package is distributed to the reliability engineer responsible for the affected system for evaluation of the significance of the plant changes on the model and the PRA conclusions. The engineer documents the evaluation using the "ECN/Drawing Evaluation Form." The entire Change Monitoring Package containing the ECNs, a list of revised drawings, and ECN/Drawing Evaluation forms are retained.

Essentially, three Phase 2 evaluation conclusions related to the impact of plant changes on the PRA are:

- *No impact, the change does not involve modeled equipment and does not need to be specifically modeled,
- *The change involves modeled equipment and decreases or insignificantly increases risk which results in conservatism in the PRA conclusions, or
- *The change involves modeled equipment and significantly increases risk which results in less conservatism in the PRA conclusions.

Changes involving modeled components will be appropriately incorporated in periodic revisions to the PRA. For the case where risk is significantly increased, the BFN project engineer will be notified.

4.0 SUMMARY OF BFN PRA CHANGES

Changes made in the BFN PRA since the January 1986 version are discussed in the following sections. These changes come from the following sources:

- °The potential refinement actions listed in Table 6.6-3 of the January 1986 PRA, and
- °Modifications identified during the iterative PRA analysis process and reviews by various organizations.

4.1 Model Refinements

Many conservatisms and assumptions in the January 1986 version were identified which, if more rigorously evaluated, could result in more realistic models and would tend to decrease the calculated overall core damage frequency. These potential PRA refinement actions, identified in Table 6.6-3 of the January 1986 version, and the extent of their incorporation into the September 1987 version are shown in Table 4-1.

Of the 56 potential refinements identified in Table 4-1, 23 have been fully or partially incorporated, 31 have been deferred, and 2 are deemed not possible at this time. Of the 31 refinements deferred, 19 require more detailed recovery analysis, 3 require more data assessment and 9 require a more realistic definition of success criteria. The 2 refinements determined not possible to incorporate at this time were so designated primarily due to their complexity. The refinements not included are not expected to significantly impact the estimated core melt frequency for BFN.

Additional refinements were made to system models, hardware data, and initiating event frequency as shown in Table 4-2. Some of the refinements listed in Table 4-2 resulted from the incorporation of changes in plant design or procedures. The basis for the system models was changed from the as-designed drawings to the as-constructed drawings, where available. System models were transferred from the Discrete Probability Distribution (DPD^R) computer code to the RISKMAN^R computer code. Other changes resulted from reviews of the PRA by various organizations and personnel internal and external to TVA. This resulted in the systems being reviewed from a new perspective, generating additional refinement actions.

4.2 Changes in Analyses

Various analyses of potential risk significant events have been completed or revised for the September 1987 version of the BFN PRA. Table 4-3 provides an overview of the status of completion of these analyses. Table 4-3 identifies which analyses were done in detail or as scoping analyses and which analyses were revised. As a minimum, a scoping analysis has been performed for these events except the loss of electrical boards below the 4KV level. For the loss of electrical boards below 4KV level, detailed dependency matrices were developed and included in the September 1987 version of the PRA.

4.3 Assessment of Operator Actions

Operator actions were assessed in the January 1986 version of the BFN PRA by assessment of the desired action, comparison with the values of similar actions in other PRAs, and engineering judgment. Assessment of the desired action involved:

- consideration of the timing of the scenario progression,
- ease of diagnosing the event (i.e., the availability of unambiguous indications and operator feedback),
- the relative stress level experienced by the operator,
- review of applicable plant procedures, and
- review of the scenarios with the plant operators.

Coupled errors, which occur when one operator error influences the likelihood that a second error will occur, were also included. Since coupled errors can introduce dependencies between top events* and since operator actions were not considered as separate top events, hardware and operator contributions to top event unavailability were combined for use in the event tree. The resulting equivalent operator error rate represents the direct human error induced failure, allowing recognition that several opportunities may exist for operator intervention and that one action may affect more than one top event.

*A top event is a system, component or operator function that determines the progression of plant responses to an initiating event.

The most significant changes to assessment of operator action between the January 1986 version and the September 1987 version of the BFN PRA are related to the evaluation of the probability that high pressure makeup failure due to operator action or inaction will lead to subsequent operator error in failure to manually depressurize the reactor vessel. Event-oriented emergency operating procedures used previously were replaced by symptom-based procedures, which significantly increase the likelihood of successful depressurization and decrease the extent of coupling of this action from manual high pressure coolant injection/reactor core isolation cooling (HPCI/RCIC) control.

Symptom-based procedures also increase attention on torus water temperature, reducing the uncoupled operator action contribution by a factor of 10 for sequences requiring establishment of torus cooling several hours into a scenario. The value for the latter action is conservative when compared with the results of a detailed formal analysis contained in a recent BWR PRA.

A more detailed description of the human factor analysis method used is provided in Appendix A.

5.0 EFFECT OF CHANGES

The changes made between the January 1986 PRA and the September 1987 version were described in Section 4.0. The effects of these changes are summarized in the following sections.

5.1 Component Failure Data and Initiating Event Frequencies

For the January 1986 version, plant specific data had been collected from the plant startup to September 1980. As identified in Table 4-2, the collection period for plant specific data has been extended from September 1980 to the end of 1985 for the September 1987 version. This resulted in:

- °A decrease in some hardware failure data values by as much as a factor of five,
- °An increase in some hardware failure data values by as much as a factor of four, and
- °No significant change in some values.

The extended collection period for plant specific data resulted in the data values being more representative of actual Browns Ferry operating history. The failure data used for the January 1986 version and for the September 1987 version are shown in Table 5-1. A second part of Table 5-1 lists the frequencies for the internal initiating events used for the January 1986 and September 1987 versions of the PRA.

As discussed in Section 4.1, the categorization of plant specific events was reviewed and revised. For example, the loss of feedwater event was divided into three subcategories. This resulted in a decrease in the importance of loss of feedwater events by allowing recovery of the feedwater system to occur. Also, the estimated plant capacity factor used to convert initiating event frequencies to calendar years in the January 1986 version was changed to the actual average plant capacity factor. These changes resulted in an overall decrease in initiating event frequencies by a factor of approximately two.

5.2 System Unavailability Impacts

Table 5-2 lists the system unavailabilities used to quantify the event trees for the January 1986 and September 1987 versions of the PRA. The unavailabilities listed are for all Common Actuation Sensors (CAS)* signals available and all electric power available (referred to as CAS state 1 and electric power state 16). The system unavailabilities for other combinations of CAS and electric power states are not necessarily the same as those listed in Table 5-2. However, the system unavailabilities for CAS state 1 and electric power state 16 are used since they give the best representation of the system unavailability (no electric power or CAS dependent failures).

The system unavailabilities differ for the January 1986 and September 1987 versions of the PRA due to the incorporation of the refinement actions listed in Tables 4-1 and 4-2. In general, the unavailabilities have decreased from the values used in the January 1986 version. In a few cases, however, the unavailabilities increased. For example, the use of plant specific data resulted in an increased unavailability for relief valves reseating (top event M1) and the use of updated design information resulted in an increase in the unavailability for recirculation pump trip (top event RP).

5.3 Core Melt Frequency (CMF) Profile

For each version of the PRA, the top 100 core damage scenarios were carefully reviewed. For the purposes of this summary report, the contribution of an event category (or system top event) to the CMF is interpreted as the "importance" of that event category. The importance of a particular event (or system top event) is defined by the sum of the frequencies of the individual top 100 sequences which contain that particular event (or system top event). Care was taken to assure that system top event importance does not include the sequences in which support system failures guarantee top event failure. This was done to prevent obscuring the true importance of the system itself. Support system importance is accounted for separately in Table 5-2.

*CAS is a BFN PRA term used to represent those sensors which provide a common accident actuation signal to multiple components and systems.

System level importance is the sum of the importance of the top events of that system. Because the event model differentiates between plant damage states in core melt scenarios (e.g., the availability of debris bed cooling is determined), the relative importance of the residual heat removal (RHR), core spray and condensate systems may be inflated. In many sequences, the success or failure of these systems does not determine whether or not a core melt occurs, rather they only affect which plant damage state is entered.

Tables 5-3 and 5-4 summarize the relative importance of the systems and internal initiating events, respectively. It should be kept in mind that the importance measures are relative. That is, the measure gives the importance of a particular initiating event or system top event in relation to the other events for that version of the PRA only. If the total CMF were the same for the two versions, then direct comparison of the relative importance values for the two version would be appropriate. However, since the CMF decreased by a factor of 10, direct comparison of the importance of an event from one version of the PRA to another could be misleading. This is because the event's relative importance may have increased while its absolute importance has decreased due to the overall decrease in CMF. For this reason, it is more appropriate to compare changes in the core melt frequency profile for each version, characterized by a collective view of the importance measures.

Table 5-3 indicates the overall relative importance of the RHR system decreased while the importance of the torus cooling mode of RHR increased. This resulted from refinement actions which decreased the requirements on the low pressure coolant injection (LPCI) mode of RHR but increased the requirements for torus cooling. The apparent large increase in the importance of HPCI/RCIC can be attributed to the redefinition of top event HP (automatic start of HPCI or RCIC). A part of this increase is also due to the inclusion of the control rod drive (CRD) system along with HPCI/RCIC in top event HP, which allows a more accurate assessment of the capability of the CRD hydraulic system (CRDHS) to provide high pressure injection. The refinement actions also increased the relative importance of emergency equipment cooling water (EECW) and the balance of plant systems. The importance of the reactor protection system (RPS), recirculation pump trip and the control air system decreased.

Table 5-4 provides an indication of the impact of the refinement actions on the importance of the initiating event categories for the January 1986 and the September 1987 versions. The refinement actions tended to increase the relative importance of transients that resulted in loss of the secondary plant systems (e.g., main steam isolation valve (MSIV) closure, loss of condenser vacuum, and loss of offsite power). At the same time, the refinements decreased the relative importance of events involving loss of control air, stuck open relief valves, and anticipated transient without scram (ATWS). This is expected since these events dominated the January 1986 core melt profile and the refinements were prioritized on the basis of impact on the dominant contributors. As indicated in Table 5-4, the core melt frequency for BFN is generally due to transients involving balance of plant systems.

Tables 5-3 and 5-4, which list the importance of systems and internal initiating events, respectively, also show that the calculated core melt frequency from the internal events has been reduced by a factor of ten through incorporation of changes between the January 1986 and the September 1987 versions.

6.0 BFN SEVERE ACCIDENT CHARACTERISTICS

The January 1986 version of the BFN PRA was known to overpredict the actual core melt frequency of the plant. Changes implemented in the BFN PRA have been focused to remove unnecessary conservatisms, incorporate plant changes in the PRA models, and refine the analyses to more closely reflect the actual plant configuration. Information gained from changes in plant design and procedures, increasing operating history, advances in analysis techniques, and PRA reviews resulted in refinement and improvement of the PRA analysis.

The September 1987 version of the BFN PRA makes use of realistic modeling assumptions and plant specific data to the greatest extent practical, with a level of detail of the analyses consistent with that found in other PRAs performed in the industry.

An overall effect of the refinements made has been to reduce the estimated CMF. Table 6-1 lists the individual plant damage state frequencies and the total CMF for the January 1986 and the September 1987 versions of the PRA. The current core melt frequency due to internal initiating events for BFN is 4.7×10^{-4} /year. The plant damage state frequencies are composed of individual event sequences grouped by common effects on the state of the plant.

The dominant sequences for the September 1987 version of the BFN PRA are identified in Table 6-2. As indicated in Table 6-2, the core melt frequency for BFN is generally due to transients involving balance of plant systems. No sequence of initiating events contributes greater than 5% to the total core melt frequency.

Therefore, it may be concluded that the September 1987 version of the BFN PRA is an accurate representation of the plant's risk and that Browns Ferry is not an outlier plant in relation to severe accident characteristics for plants of similar type and vintage.

APPENDIX A

ASSESSMENT OF OPERATOR ACTIONS METHODOLOGY

Care was taken in acknowledging the possibility of "coupled" operator errors occurring. Coupled errors occur when one error influences the likelihood that a second error occurs. Such errors may introduce dependencies between several top events. Such coupled errors were included in the BFN PRA. Because operator actions were not identified as separate top events in the PRA, it was necessary to combine hardware and operator contributions to top event unavailability for use in the event tree.

An example of how this coupling was handled is the assessment of operator errors involving manual control of the injection systems and manual depressurization. High pressure vessel makeup by the high pressure coolant injection (HPCI) or reactor core isolation cooling (RCIC) system in transient events is defined as top event HP. Manual depressurization is defined as top event V2.

For most transients, the value for top event HP that is used in the event tree quantification is given by

$$HP_T = HP_H + HP_O - HP_H HP_O$$

The H and O subscripts refer to the hardware and operator contributors to unavailability, respectively. Both HPCI and RCIC receive an automatic start signal on level 2 and a trip signal on level 8. Following a high level trip, HPCI has the potential to continue to automatically cycle; hardware failures (e.g., HPCI pump fails to start on demand) are possible during cycling. A HPCI/RCIC cycling model was developed which considers the operator as an integral part of the process. Critical timing, alarms and instrument information are a function of initial vessel refill rate (e.g., whether HPCI is available) and the location within the cycle. The resulting value for HP_O is interpreted as an equivalent operator error rate and represents the direct human error induced failure of HPCI/RCIC recognizing that several opportunities exist for the operator to take control of HPCI/RCIC flow.

The expression for the closely related top event V2 for the case where HP has failed is given by:

$$V2 = [(1-HP_H)HP_O[V2_H + (1-V2_H)V2_O^*] + HP_H [(1-V2_H)V2_O + V2_H]]/HP_T$$

The parameter $V2_O^*$ is interpreted as the probability that the operator fails to manually depressurize the vessel after high pressure makeup (HPCI and RCIC) fails due to operator action or inaction. In this manner, the potential coupling of operator errors between top events HP and V2 is explicitly accounted for in the quantification of the event trees.

Status of Potential Refinements Identified in
Table 6.6-3 of January 1986 PRA

<u>Designator</u> <u>from Table 6.6-3</u>		<u>Refinement⁽¹⁾</u> <u>Actions</u>	<u>Comment</u>
A	Modify ATWS model to reflect ORNL/SASA results on high pressure injection success criteria	Partial	High Pressure makeup with RCIC and CRDHS added. The ATWS model is still conservative.
B	Refine high pressure makeup used in ATWS model	Partial	Iterative refinements made in the operator response model.
C	Refine RPS Top event definitions to include additional signals	Incorporated	Added second input trip signal to RPS top events.
D	Include ATWS sequences for 1-3 Stuck Open Relief Valves (SORVs) with scram failure	Incorporated	In the original event model, sequences went directly to core melt for 1-3 SORVs and RPS failure.
E	Include manual actions to start RHR pumps for torus cooling (RI) and refine human error model for 1-3 SORVs	Incorporated	Replaced top event RI with R7. Top event R7 includes manual start. RI does not.
F	Includes manual start of RHR (and refine human error model) for multiunit events (Also consider: 1 RHR pump and 1 RHR Service Water (RHRSW) pump with 1/3 flow)	Incorporated	Added manual start to top event RD.
G	Reconsider model of diesel generator failure during operation	Deferred	Involved changing diesel generator failure rate to time dependent failure. Importance of this refinement action deemed small. EECW dominates station blackout frequency.

Status of Potential Refinements Identified in
Table 6.6-3 of January 1986 PRA

<u>Designator from Table 6.6-3</u>		<u>Refinement⁽¹⁾ Actions</u>	<u>Comment</u>
H	Include recovery from Standby Liquid Control (SLC) SI (ATWS)	Deferred	Importance of this refinement action decreased with the incorporation of other ATWS related refinements.
I	Manually open shutdown cooling valves	Incorporated	Based on actual plant experience.
J	Recover air (e.g., use bottled air)	Deferred	Importance of this refinement action decreased with the incorporation of other control air system refinements.
K	Include CRDHS and multiunit equivalent of manual start of RHR pumps for torus cooling (R7) for loss of control air and loss of offsite power events (LOSP)	Not Possible	Refinement action complex to incorporate; on LOSP, no power to CRDHS pumps. Loss of Control Air Model conservatively envelops loss of Raw Cooling Water (RCW) and loss of Reactor Building Closed Cooling Water (RBCCW).
L	Consider transient recovery (i.e., return to power) for successful blowdown cases	Deferred	Likelihood of successful return to power not evaluated. No experience base available.
M	Include CRDHS for sequences with successful manual depressurization; low pressure makeup; and, containment cooling	Not Possible	Complex to incorporate due to CRDHS dependencies. Would have to be done on a sequence by sequence basis.
N	Recategorize relief valve data (transient/1-3 SORVs)	Deferred	Analysis required to differentiate plant response for differing number of relief valves stuck open.

Status of Potential Refinements Identified in
Table 6.6-3 of January 1986 PRA

Designator from Table 6.6-3		Refinement ⁽¹⁾ Actions	Comment
O	Restructure transient events to reflect number of relief valves actually lifted	Deferred	Refinement action complex to incorporate. Would require detailed analysis to identify number of valves lifting for different sequences.
P	Manual action to start HPCI and EECW (H4, EE) buy time to manually open RHR injection valves (e.g., transient/loss of coolant accident (LOCAs) [See Q]	Incorporated	Action Q is a parallel path to action P.
Q	Manual action to start HPCI and EECW (H4 EE); include CRDHS and manual start of RHR for torus cooling (e.g., transient/LOCAs) [See P]	Deferred	Incorporation of Action P reduced benefit of this action.
R	Reanalyze IE data (1 SORV versus 2-3 SORVs; exclude electromatic and 3-stage valves)	Deferred	Existing data considered conservative. Two-stage Target Rock data was used in the Safety Relief Valve (SRV) systems analysis in the January 1986 version; initiating event prior population data contains data from all US BWRs.
S	Manual start of HPCI and EECW (H3, E2); buy time to open RHR injection valves (small LOCA) [see T]	Incorporated	Action T is a parallel path to action S.
T	Manual start of HPCI and EECW (H3, E2); and manual start of RHR for torus cooling (small LOCA) [see S]	Deferred	Incorporation of Action S reduced benefit of this action.
U	Manual start of HPCI and EECW (HP, EE); buy time to open RHR injection valves (transients) [see V]	Incorporated	Action V is a parallel path to action U.

Status of Potential Refinements Identified in
Table 6.6-3 of January 1986 PRA

<u>Designator from Table 6.6.3</u>		<u>Refinement⁽¹⁾ Actions</u>	<u>Comment</u>
V	Manual start of HPCI and EECW (HP, EE); and RHR pumps for torus cooling in model transients [see U]	Deferred	Incorporation of Action U reduced benefit of this action.
W	Resort Loss of Feedwater (LOF) IE data	Incorporated	Loss of feedwater broken up into three categories; total recoverable, total nonrecoverable, and partial loss. Previously, any LOF (complete or partial) was assumed to be total loss.
X	Replace RHR top RB with R7 (re-examine human error model; small LOCA)	Incorporated	Originally, top RB required auto-start of RHR pumps and manual alignment to torus cooling. Replaced with R7, manual start of torus cooling.
Y	Reconsider operator action portion of manual depressurization for cases other than HPCI and RCIC initial failure (see GG, AF)	Incorporated	Refined operator error portion of HP top event model.
Z	Include transient recovery after 1 HPCI/RCIC vessel fill after pressure regulator fails closed	Deferred	Requires analysis to determine likelihood of successful recovery.
AA	Include transient recovery after 1 HPCI/RCIC vessel fill after MSIV closure	Deferred	See Z.
BB	Include transient recovery after 1 HPCI/RCIC vessel fill after loss of feedwater	Deferred	See Z.

Status of Potential Refinements Identified in
Table 6.6-3 of January 1986 PRA

<u>Designator from Table 6.6-3</u>		<u>Refinement⁽¹⁾ Actions</u>	<u>Comment</u>
CC	Reconsider MSIV closure initiating event data (single valve closure events below a certain power level would not result in closure of all valves)	Deferred	Differentiation of initial power levels deemed to not be beneficial compared with complexity to incorporate refinement.
DD	Include transient recovery after HPCI/RCIC vessel fill after loss of condenser vacuum	Deferred	See Z.
EE	Reanalyze initial control of feedwater pumps (FA and FB)	Deferred	Current model is conservative.
FF	Reconsider model coupling operator action involved with feedwater, HPCI, RCIC Control and manual depressurization	Incorporated	Revised model coupling human error.
GG	Reconsider conditional human error associated with manual depressurization for feedwater rampup sequences	Deferred	Incorporation of action Y decreased importance of this action.
HH	Recover 480-volt board IA	Incorporated	Incorporated on sequence specific basis.
II	Recover 4-kV shutdown board from relay test	Incorporated	Incorporated on sequence specific basis.
JJ	Recover 480-volt board IA before lack of room cooling fails pumps	Deferred	Analysis required to determine time available till RHR pumps fail.
KK	Recover 4-kV shutdown board IA before lack of room cooling fails pumps after LOSP	Deferred	See JJ.

Status of Potential Refinements Identified in
Table 6.6-3 of January 1986 PRA

<u>Designator from Table 6.6-3</u>		<u>Refinement⁽¹⁾ Actions</u>	<u>Comment</u>
LL	Remove relief valve reseating top (MI) from ATWS model	Incorporated	Number of stuck open relief valves has no impact on plant response to ATWS based on ORNL analysis.
MM	Reconsider model coupling operator action involved with HPCI and RCIC control and manual depressurization coupling model	Incorporated	Recent model accounts for availability of new symptom based Emergency Operating Procedures (EOP's).
NN	Recover 480-volt board IA before lack of room cooling fails pumps after LOSP	Deferred	See JJ.
OO	Recover 4-kV shutdown board before lack of room cooling fails pumps after LOSP	Deferred	See JJ.
PP	Recover power after 1 fill by HPCI/RCIC after LOSP	Deferred	Analysis required to determine boiloff time from level 8 rather than level 2.
QQ	Reconsider diesel generator common cause model; incorporate single and double diesel generator recovery into electric power system model	Partial	Updated common cause parameters utilized in September 1987 version.
RR	Reformulate model to take credit for 1 core spray pump supplying makeup after successful operation of HPCI, RCIC; include top event RD and 480-V board recovery after LOSP	Deferred	Analysis required to verify success criteria.

Status of Potential Refinements Identified in
Table 6.6-3 of January 1986 PRA

<u>Designator from Table 6.6-3</u>	<u>Refinement⁽¹⁾ Actions</u>	<u>Comment</u>
RR	Like action RR except RHR flow is split to provide both long-term makeup as well as torus cooling after LOSP	Deferred See RR.
RT	Use existing RHR pumps to reflood; cycle as necessary, use 1 core spray pump to maintain level after LOSP	Deferred See RR.
RU	Include multiple (up to 3) diesel generator recovery; reconsider common cause model	Partial See QQ Above
RV	Include multiple (up to 4) diesel generator recovery; reconsider common cause model	Partial See QQ Above
WW	Re-examine MSIV closure (GI); develop data for MSIVs	Incorporated Initial versions of PRA used generic valve data for MSIVs. September 1987 version used MSIV specific data.
XX	Treat MSIV closure failure like large LOCA; use low pressure emergency cooling system (LPECS); must consider long-term inventory makeup	Deferred Importance of this refinement action decreased with the incorporation of action WW.
YY	Include manual/alternative actions to close MSIVs/Bypass valves and turbine control valves (TCV) or turbine stop valves (TSVs) to terminate transients	Deferred Importance of this refinement action decreased with the incorporation of action WW.
AB	Recover RHR or use other unit RHR to cool torus	Deferred Requires multiunit modeling to determine availability of other unit RHR pumps.

Status of Potential Refinements Identified in
Table 6.6-3 of January 1986 PRA

<u>Designator</u> <u>from Table 6.6-3</u>		<u>Refinement⁽¹⁾</u> <u>Actions</u>	<u>Comment</u>
AC	Reanalyze operator action associated with IPCC with timing considerations specific to 1 or 2-S SORVs	Deferred	Cannot be incorporated until action O and R are incorporated.
AD	Eliminate RHR for containment spray (SP) requirement	Incorporated	Requirement for containment spray for large break events deleted based on ORNL analysis.
AE	Include RHR (RB) and 2 loops of core spray as a success sequence for large LOCA	Deferred	Requires analysis to verify success criteria.
AF	Reconsider operator action associated with manual depressurization for feedwater rampup sequences	Deferred	See GG.

Note (1):

These actions represent a combination of refinements performed in the iterative revision process between the January 1986 and the September 1987 versions of the BFN PRA.

Note (2):

Incorporated: Refinement Action Incorporated

Partial: Refinement Action Partially Incorporated

Deferred: Refinement Action Deferred

Not Possible: Refinement Action would require considerable modeling effort or not possible

ADDITIONAL REFINEMENTS MADE IN THE PRA

I. SYSTEM MODEL CHANGES

<u>System</u>	<u>Additional Refinements</u> (1)
Common Actuation Sensors	Eliminated common cause gamma model; incorporated Analog Trip System (ATS) modification
Residual Heat Removal Service Water	Revised maintenance model to reflect actual EECW/RHREW pump maintenance practices
Raw Cooling Water	Removed requirement for Auxiliary Raw Cooling Water System
Emergency Equipment Cooling Water	Included recovery from logic system surveillance test; manual start of pumps included
Recirculation Pump Trip	Upgraded to reflect latest design information and ATS
Condensate	Cycling/startup bypass valve model modified to more correctly represent actual case
Drywell control Air	Interconnection with plant air included, minor corrections to equations
Electric power System	Recovery Expanded for Offsite Power and 480v ac Shutdown Boards 1A and 1B; models for 250v dc RMOV board 1B and Battery Board 3 revised for LOSEP events.
Reactor Protection System	Second input signals added for top events T3 and T6 to reduce dependence on input signals; incorporated ATS modification
Residual Heat Removal	Minor corrections to equations
Primary Containment Isolation	Incorporated ATS modification; model requantified utilizing MSIV specific data

ADDITIONAL REFINEMENTS MADE IN THE PRA

II. EVENT MODEL CHANGES

	<u>Additional Refinements</u> (1)
Transient Events involving ADS	Models revised reflect new EOIs; ADS replaced by manual depressurization
Transient Events involving EPS 1	Expanded use of early manual initiation of RHR and core spray for sequences in CAS state 1 or 2 and with early depressurization
Transient Events	Refinement of HPCI/RCIC cycling model; CRD included where appropriate
Loss of feedwater	Models developed for three subcategories of loss of feedwater
MSIV Closure; Loss of Condenser Vacuum; Pressure Regulator Failure-closed; loss of Plant Air; loss of feedwater	Figures of separate event trees without recovery deleted from document

III. COMPONENT FAILURE DATA CHANGES

<u>Component</u>	<u>Additional Refinements</u>
HPCI pump	Data base updated to reflect system modifications
RCIC pump	Collection period extended for plant specific data
Core Spray pumps	Collection period extended for plant specific data
RHR pumps	Collection period extended for plant specific data
Standby Liquid Control pumps	Collection period extended for plant specific data
RHRSW/EECW pumps	Collection period extended for plant specific data
Feedwater pumps	Collection period extended for plant specific data
Condensate pumps	Collection period extended for plant specific data

ADDITIONAL REFINEMENTS MADE IN THE PRA

III. COMPONENT FAILURE DATA CHANGES (Continued)

<u>Component</u>	<u>Additional Refinements</u>
Condensate Booster pumps	Collection period extended for plant specific data
Motor Operated valves	Collection period extended for plant specific data
Manual valves	Data for failure mode "failure to open on demand" added
Main Steam Relief Valves	Collection period extended for plant specific data
Main Steam Isolation Valves	Collection period extended for plant specific data; Data for failure mode "failure to close on demand" added
RHR Heat Exchangers	Collection period extended for plant specific data
RHR room cooler	Collection period extended for plant specific data
CS room cooler	Collection period extended for plant specific data
Air compressors	Collection period extended for plant specific data
Control Air dryers	Plant data reviewed and refined
Air Relief Valves	Data for 1/2" to 2" spring actuated valves "leaking sticking open or lifting prematurely" added

IV. INITIATING EVENT FREQUENCY DATA CHANGES

All Internal Initiating Event Categories	(1) Collection period extended for plant specific data. (2) Categorization of all plant specific events reviewed and revised as necessary. (3) Capacity factor used to express frequency in terms of calendar years revised.
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Note (1): These actions represent a combination of refinements performed between the January 1986 version and the September 1987 version of the BFN PRA.

TABLE 4-3

OVERVIEW OF CHANGES IN ANALYSES

	January 1986 Version	September 1987 Version
Internal Events as Described in Section 6.4 of the PRA	A	AR
Other Events as Described in Section 6.5 of the PRA:		
Loss of Electrical Boards	N	N
Loss of HVAC	N	S
Common Instrument Tap Considerations	N	A
Interfacing LCCAs	N	A
Multiple Unit Interactions	N	S
Loss of the Condensate Storage Tank	N	S
Torus Rupture	S	A
MSRV Tailpipe Vacuum Breaker Stuck Open	N	S
Scram Discharge Volume Break	N	A
Loss of Decay Heat Removal	N	S
Loss of Reactor Building Closed Cooling Water	N	S
Purging During Operation	N	S
Inadvertent Fire Suppression System Operation	N	A
Loss of Recirculation Pump Seal Cooling	N	S
Loss of Flow Through Traveling Screens	N	S
Common Accident Signal Consideration	N	S
Radiological Releases by Means Other Than the Core	N	A
External Events as Described in Section 8 and Appendix E of the PRA		
Earthquake	A	AR
Extreme Winds and Tornado	A	A
Aircraft Impacts	A	A
Fires	A	AR
Turbine Missiles	S	S
Flooding (External)	A	A
Flooding (Internal)	A	AR
Transportation Accidents	S	S
Toxic Gas Release	S	S
Containment Response Model	S	S
Offsite Consequence Analysis	S	S

A: Analysis Performed

AR: Analysis Revised

S: Scoping Analysis Performed

N: Analysis Not Performed

IMPACT OF CHANGES ON COMPONENT FAILURE DATE
INITIATING EVENT FREQUENCIES

I. COMPONENT FAILURE DATA

All values shown are the mean of a distribution and given as occurrences per demand or per hour, as noted.

<u>Component/Failure Mode</u>	<u>January 1986 Version</u>	<u>September 1987 Version</u>
<u>HPCI pump</u>		
failure to start on demand	6.62×10^{-2}	2.41×10^{-2}
failure during operation (per hour)	3.95×10^{-4}	3.87×10^{-4}
<u>RCIC pump</u>		
failure to start on demand	6.06×10^{-2}	3.98×10^{-2}
failure during operation (per hour)	3.95×10^{-4}	3.87×10^{-4}
<u>Core Spray pumps</u>		
failure to start on demand	1.26×10^{-3}	9.72×10^{-4}
failure during operation (per hour)	2.62×10^{-5}	2.62×10^{-5}
<u>RHR pumps</u>		
failure to start on demand	1.6×10^{-3}	1.83×10^{-3}
failure during operation (per hour)	1.76×10^{-5}	1.41×10^{-5}
<u>Standby Liquid Control pumps</u>		
failure to start on demand	2.27×10^{-3}	1.85×10^{-3}
failure during operation (per hour)	2.62×10^{-5}	2.61×10^{-5}
<u>KHRSW/EECW pumps</u>		
failure to start on demand	3.44×10^{-3}	6.25×10^{-4}
failure during operation (per hour)	6.47×10^{-5}	4.69×10^{-5}
<u>Feedwater pumps</u>		
failure during operation (per hour)	2.85×10^{-5}	4.99×10^{-5}
<u>Condensate pumps</u>		
failure during operation (per hour)	1.97×10^{-5}	1.55×10^{-5}
<u>Condensate Booster Pumps</u>		
failure during operation (per hour)	2.64×10^{-5}	2.21×10^{-5}
<u>Motor Operated Valves</u>		
failure to operate on demand	3.28×10^{-3}	2.45×10^{-3}

IMPACT OF CHANGES ON COMPONENT FAILURE DATE
INITIATING EVENT FREQUENCIES

<u>Component/Failure Mode</u>	<u>January 1986 Version</u>	<u>September 1987 Version</u>
<u>Manual Valves</u>		
failure to open on demand	--	2.13×10^{-4}
<u>Main Steam Relief Valves</u>		
failure to open on demand	3.63×10^{-3}	4.05×10^{-3}
failure to reseal on demand	2.63×10^{-3}	5.22×10^{-3}
<u>Main Steam Isolation Valves</u>		
failure to open on demand	2.80×10^{-4}	1.93×10^{-4}
failure to close on demand	-- *	3.77×10^{-5}
failure during operation - transfers closed (per hour)	3.63×10^{-6}	2.07×10^{-6}
<u>RHR Heat Exchangers</u>		
failure during operation - excessive leakage or fouling (per hour)	1.15×10^{-6}	1.02×10^{-6}
<u>RHR Room Coolers</u>		
failure to operate on demand	1.45×10^{-3}	1.31×10^{-3}
failure during operation due to excessive leaking or fouling (per hour)	2.69×10^{-6}	1.63×10^{-6}
<u>CS Room Coolers</u>		
failure to operate on demand	1.21×10^{-3}	4.41×10^{-3}
failure during operation due to excessive leaking or fouling (per hour)	2.69×10^{-6}	2.69×10^{-6}
<u>Drywell Air Compressors</u>		
failure during operation (per hour)	7.40×10^{-5}	8.89×10^{-5}
<u>Plant Control Air Compressors</u>		
failure during operation (per hour)	7.34×10^{-5}	8.96×10^{-5}
<u>Control Air Dryers</u>		
failure during operation (per hour)	3.55×10^{-5}	3.34×10^{-5}
<u>Air Relief Valves</u>		
failure during plant operation: lifting prematurely (per hour)	--	6.51×10^{-6}

*In the January 1986 version, data for motor operated valve failure to operate on demand (mean value: 3.28×10^{-3}) was used.

IMPACT OF CHANGES ON COMPONENT FAILURE DATA AND
INITIATING EVENT FREQUENCIES

II. INITIATING EVENT DATA

All values shown are the mean of a distribution.

Internal Initiations Event Category (event No.)	Annual Frequency (per calendar year)	
	January 1986 Version	September 1987 Version
Feedwater Rampup (1)	2.26×10^{-1}	1.23×10^{-1}
Moderator Temperature Decrease (2)	3.32×10^{-1}	1.89×10^{-1}
MSIV Closure (3)	9.54×10^{-1}	7.50×10^{-1}
Loss of Condenser Vacuum (4)	3.47×10^{-1}	2.36×10^{-1}
Loss of Offsite Power (5)	6.10×10^{-2}	3.86×10^{-2}
Pressure Regulator Failure-Closed (6A)	3.91×10^{-1}	2.82×10^{-1}
Pressure Regulator Failure-Open (6B)	6.35×10^{-2}	4.16×10^{-2}
Other Turbine Trip (7)	2.93	1.68
Loss of Feedwater* (8)	1.08	--
Loss of Feedwater-Not Immediately Restorable* (8A)	--	4.93×10^{-1}
Loss of Feedwater-Immediately Restorable* (8B)	--	1.20×10^{-1}
Partial Loss of Feedwater* (8C)	--	1.48×10^{-1}
Loss of Control Air (9)	1.56×10^1	9.66×10^{-2}
Other Scram (10)	4.16	3.40
Main Steam Line Break-Outside Containment (11A)	6.16×10^{-5}	3.32×10^{-5}
Main Steam Line Break-Inside Containment (11B)	7.15×10^{-5}	3.86×10^{-5}
Feedwater Line Break-Outside Containment (12A)	9.40×10^{-5}	5.07×10^{-5}
Feedwater Line Break-Inside Containment (12B)	4.36×10^{-5}	2.35×10^{-5}
HPCI Steamline Break (13)	3.30×10^{-5}	1.78×10^{-5}
RWCU Break-Return Line (14A)	7.18×10^{-5}	3.87×10^{-5}
RWCU Break-Suction Line (14B)	1.08×10^{-4}	5.80×10^{-5}
RCIC Steamline Break (15)	4.29×10^{-5}	2.32×10^{-5}
Recirculation Discharge Line Break (16)	3.07×10^{-4}	1.65×10^{-4}
Recirculation Suction Line Break (17)	9.13×10^{-5}	4.91×10^{-5}
Core Spray Line Break-Inside Containment (18)	8.21×10^{-5}	4.43×10^{-5}

IMPACT OF CHANGES ON COMPONENT FAILURE DATA AND
INITIATING EVENT FREQUENCIES

Internal Initiations Event Category (event No.)	Annual Frequency (per calendar year)	
	January 1986 Version	September 1987 Version
Medium Steam Break I - Inside Containment (19A)	6.86×10^{-6}	3.70×10^{-6}
Medium Steam Break II - Inside Containment (19B)	7.46×10^{-6}	4.03×10^{-6}
Medium Steam Break - Inside Containment (19C)	4.53×10^{-5}	2.44×10^{-5}
Small Steam Break - Inside Containment (20A)	2.42×10^{-3}	1.71×10^{-3}
Small Water Break - Inside Containment (20B)	4.04×10^{-2}	2.53×10^{-2}
Inadvertent Opening of 1-3 MSRVs (21A)	1.13×10^{-1}	6.58×10^{-2}
Inadvertent Opening of 4 or more MSRVs (21B)	1.96×10^{-3}	1.41×10^{-3}

*Loss of Feedwater divided into three subcategories for the September 1987 version.

IMPACT OF CHANGES ON SYSTEM UNAVAILABILITIES

<u>System/Top Event</u>	January 1986 ⁽¹⁾ <u>Version</u>	September 1987 ⁽¹⁾ <u>Version</u>
<u>Residual Heat Removal System</u>		
R1: Auto start of 2 RHR pumps/torus cooling	9.20×10^{-4}	2.82×10^{-4}
R2: Auto start of 3 RHR pumps/torus cooling and injection	1.83×10^{-2}	1.34×10^{-2}
R5: Auto start of 2 RHR pumps (different loops)/torus cooling (High Drywell Pressure)	3.98×10^{-2}	3.32×10^{-2}
R7: Torus cooling	2.82×10^{-4}	1.03×10^{-4}
RB: Auto start of 2 RHR pumps/torus cooling, (High Drywell Pressure)	1.96×10^{-3}	Note 2
RD: Auto start of 2 RHR pumps/torus cooling (multiunit)	1.96×10^{-3}	1.03×10^{-3}
SD: Shutdown cooling	1.93×10^{-2}	1.99×10^{-2}
SP: Containment spray	9.90×10^{-3}	Note 3
<u>Core Spray System</u>		
C1: Auto start of 1 core spray loop (High Drywell Pressure)	5.29×10^{-4}	6.01×10^{-4}
CS: Auto start of 1 core spray loop	5.29×10^{-4}	6.0×10^{-4}
<u>HPCI/RCIC*</u>		
H3: Auto start of HPCI or RCIC (High Drywell Pressure)	1.48×10^{-2}	9.57×10^{-3}
H4: Auto start of HPCI	1.15×10^{-1}	1.06×10^{-1}
HP: Auto start of HPCI or RCIC	1.23×10^{-2}	8.15×10^{-3}
<u>Emergency Equipment Cooling Water System</u>		
EE: Auto start of 2 pumps	2.42×10^{-3}	6.41×10^{-4}
<u>Main Steam/Condensate/Feedwater</u>		
C0: Condensate/condensate booster pumps	1.45×10^{-3}	1.18×10^{-3}
F1: Manual reestablishment of feedwater	1.49×10^{-3}	3.45×10^{-4}
FT: Feedwater Pump Trip	7.96×10^{-3}	7.83×10^{-3}
EH: Turbine Control System	1.67×10^{-2}	1.79×10^{-2}

IMPACT OF CHANGES ON SYSTEM UNAVAILABILITIES

<u>System/Top Event</u>	January 1986 ⁽¹⁾ <u>Version</u>	September 1987 ⁽¹⁾ <u>Version</u>
<u>Depressurization/Relief Valves</u>		
V1: Manual Actuation of one Relief Valve	6.32×10^{-4}	1.88×10^{-4}
V2: Manual Depressurization	1.40×10^{-3}	3.31×10^{-4}
M1: Relief Valves Reseat	3.41×10^{-2}	6.69×10^{-2}
<u>Reactor Protection System</u>		
T2: Scram (L3 or High Drywell Pressure)	1.25×10^{-4}	1.12×10^{-4}
T3: Scram (L3 or MSIV position)	4.47×10^{-4}	1.12×10^{-4}
<u>Recirculation Pump Trip</u>		
RP: Recirculation Pump Trip	1.41×10^{-4}	6.51×10^{-3}
<u>Raw Cooling Water</u>		
CW: Raw Cooling Water	2.96×10^{-4}	1.12×10^{-4}
<u>Electric Power System</u>		
EPS16: Unit 1 and 2 4KV shutdown boards available	9.92×10^{-1}	9.97×10^{-1}
EPS1: 4KV shutdown board A unavailable	5.95×10^{-4}	7.83×10^{-4}
EPS2: 4KV shutdown board B unavailable	5.95×10^{-4}	7.85×10^{-4}
EPS3: 4KV shutdown board C unavailable	5.91×10^{-4}	6.46×10^{-4}
EPS4: 4KV shutdown board D unavailable	5.91×10^{-4}	9.79×10^{-4}
EPS5: 4KV shutdown boards A B unavailable	6.06×10^{-6}	1.52×10^{-5}
EPS6: 4KV shutdown boards A C unavailable	7.04×10^{-7}	2.97×10^{-7}
EPS7: 4KV shutdown boards A D unavailable	7.04×10^{-7}	6.31×10^{-7}
EPS8: 4KV shutdown boards B C unavailable	7.04×10^{-7}	5.72×10^{-7}
EPS9: 4KV shutdown boards B D unavailable	7.04×10^{-7}	6.31×10^{-7}
EPS10: 4KV shutdown boards C D unavailable	1.31×10^{-4}	2.80×10^{-6}
EPS11: 4KV shutdown boards A B C unavailable	2.10×10^{-7}	3.41×10^{-7}
EPS12: 4KV shutdown boards A B D unavailable	2.10×10^{-7}	3.44×10^{-7}
EPS13: 4KV shutdown boards A C D unavailable	4.13×10^{-9}	5.91×10^{-10}
EPS14: 4KV shutdown boards B C D unavailable	4.13×10^{-9}	5.91×10^{-10}
EPS15: 4KV shutdown boards A B C D unavailable	3.10×10^{-9}	1.57×10^{-10}
5A: Early Recovery of Offsite Power	4.50×10^{-2}	4.50×10^{-2}
5B: Backfeed of Unit Board	3.00×10^{-1}	3.00×10^{-1}

IMPACT OF CHANGES ON SYSTEM UNAVAILABILITIES

<u>System/Top Event</u>	January 1986 ⁽¹⁾ <u>Version</u>	September 1987 ⁽¹⁾ <u>Version</u>
<u>Primary Containment Isolation</u>		
GI: MSIV Isolation-L2 or low steamline Pressure	5.04×10^{-4}	1.50×10^{-4}
GH: MSIV Isolation-L2	7.93×10^{-4}	1.55×10^{-4}
<u>Plant Air</u>		
AI: Plant Air	6.11×10^{-3}	1.08×10^{-3}
<u>Standby Liquid Control</u>		
SL: Standby Liquid Control	1.80×10^{-2}	1.81×10^{-2}
<u>Common Actuation Sensors**</u>		
All sensors Available	9.74×10^{-1}	9.79×10^{-1}
Minimum sensors available	2.53×10^{-2}	2.06×10^{-2}
Minimum sensors unavailable	8.12×10^{-4}	4.89×10^{-4}

*In the September 1987 version, credit is taken for the CRD system where appropriate. For the earlier versions, CRD was modeled as a separate top event. This caused the apparent increase in the HPCI/RCIC unavailabilities for the September 1987 version.

**CAS unavailabilities are for the alpha model.

Notes

1. System unavailabilities are for CAS state 1 and electric power state 16, offsite power available.
2. Top event RB was replaced with top event R7.
3. Top event SP was deleted between the January 1986 version and the September 1987 version.

CHANGES TO SYSTEM IMPORTANCE

<u>System/Top Event</u>	<u>January 1986 Version</u>	<u>September 1987 Version</u>
<u>Residual Heat Removal System</u>	<u>26.55%</u>	<u>17.53%</u>
R1: Auto start of 2 RHR pumps/torus cooling	10.28	--
R2: Auto start of 3 RHR pumps/torus cooling and injection	--	0.51
R5: Auto start of 2 RHR pumps (different loops)/torus cooling (High Drywell Pressure)	--	0.27
R7: Torus cooling	3.88	14.86
RB: Auto start of 2 RHR pumps/torus cooling (High Drywell Pressure)	3.28	--
RD: Auto start of 2 RHR pumps/torus cooling (multiunit)	3.45	1.71
SD: Shutdown cooling	5.00	0.18
SP: Containment spray	0.66	--
<u>Core Spray</u>	<u>--</u>	<u>1.97%</u>
C1: Auto start of 1 core spray loop (High Drywell Pressure)	--	0.91
CS: Auto start of 1 core spray loop	--	1.06
<u>HPCI/RCIC*</u>	<u>23.08%</u>	<u>30.24%</u>
H3: Auto start of HPCI or RCIC (High Drywell Pressure)	--	0.51
H4: Auto start of HPCI	6.47	1.29
HP: Auto start of HPCI or RCIC	16.61	28.44
<u>Emergency Equipment Cooling Water</u>	<u>0.36%</u>	<u>5.12%</u>
EE: Auto start of 2 pumps	0.36	5.12
<u>Main Steam/Condensate/Feedwater</u>	<u>3.73%</u>	<u>5.53%</u>
CO: Condensate/Condensate booster pumps	1.21	2.89
F1: Manual reestablishment of feedwater	2.52	--
FT: Feedwater Pump Trip	--	2.13
EH: Turbine Control System	--	0.51

TABLE 5-3

CHANGES TO SYSTEM IMPORTANCE

<u>System/Top Event</u>	<u>January 1986 Version</u>	<u>September 1987 Version</u>
<u>Depressurization/Relief Valves</u>	<u>16.78%</u>	<u>25.36%</u>
V1: Manual Actuation of One Relief Valve	--	--
V2: Manual Depressurization	15.60	25.36
M1: Relief Valves Reseat	1.18	--
<u>Reactor Protection System</u>	<u>1.44%</u>	<u>0.46%</u>
T2: Scram (L3 or High Drywell Pressure)	--	0.46
T3: Scram (L3 or MSIV position)**	1.44	--
<u>Recirculation Pump Trip</u>	<u>--</u>	<u>0.44%</u>
RP: Recirculation Pump Trip Function	--	0.44
<u>Raw Cooling Water</u>	<u>0.31%</u>	<u>--</u>
CW: Raw Cooling Water	0.31	--
<u>Electric Power</u>	<u>70.77%</u>	<u>76.29%</u>
EPS16 Unit 1 and 2 4KV Shutdown Boards available	54.39	51.31
EPS1 4KV Shutdown Board A unavailable	7.40	7.06
EPS2 4KV Shutdown Board B unavailable	--	0.25
EPS3 4KV Shutdown Board C unavailable	0.28	4.81
EPS4 4KV Shutdown Board D unavailable	--	--
EPS5 4KV Shutdown Boards A B unavailable	1.33	4.35
EPS6 4KV Shutdown Boards A C unavailable	1.28	3.07
EPS7 4KV Shutdown Boards A D unavailable	1.44	0.96
EPS8 4KV Shutdown Boards B C unavailable	1.36	3.47
EPS9 4KV Shutdown Boards B D unavailable	1.25	0.82
EPS10 4KV Shutdown Boards C D unavailable	0.22	--
EPS11 4KV Shutdown Boards A B C unavailable	0.31	0.19
EPS12 4KV Shutdown Boards A B D unavailable	0.48	--
EPS13 4KV Shutdown Boards A D C unavailable	0.46	--
EPS14 4KV Shutdown Boards B C D unavailable	0.30	--
EPS15 4KV Shutdown Boards A B C D unavailable	0.27	--
5A: Early Recovery of Offsite Power	7.28	7.23
5B: Backfeed of Unit Board	--	0.43

CHANGES TO SYSTEM IMPORTANCE

<u>System/Top Event</u>	<u>January 1986 Version</u>	<u>September 1987 Version</u>
Primary Containment Isolation	0.92%	1.65%
GI: MSIV Isolation (L2 or Low Steamline Pressure)	0.92	1.32
GH: MSIV Isolation (L2)	--	0.33
Plant Air	2.06%	--
AI: Plant Air	2.06	--
Standby Liquid Control	0.52%	1.79%
SL: Standby Liquid Control	0.51	1.79
Common Actuation Sensors	70.87%	75.29%
All Sensors Available	61.51	72.23
Minimum Sensors Available	--	--
Minimum Sensors Not Available	9.36	3.06
Percent CMF Represented by top 100 Sequences	70.87%	76.29%
Absolute CMF from Internal Events (per calendar year)	3.9×10^{-3}	4.7×10^{-4}

*In the September 1987 version, credit is taken for the CRD System where appropriate.

**Second input signal added between the January 1986 version and the September 1987 version.

General Notes:

See section 5.3 for definition of importance.

Standby Gas Treatment System not included (SBGT only determined plant damage state).

Internal events or y.

Importance values less than 0.01 are indicated by "--"

IMPORTANCE OF INTERNAL INITIATING EVENTS

Internal Initiating Event Category (event no.)	January 1986 Version (%)	September 1987 Version (%)
Feedwater Rampup (1)	1.30	1.12
Moderator Temperature Decrease (2)	--	--
MSIV Closure (3)	3.62	8.01
Loss of Condenser Vacuum (4)	1.51	3.12
Loss of Offsite Power (5)	11.10	12.36
Pressure Regulator Failure - Closed (6A)	1.95	3.90
Pressure Regulator Failure - Open (6B)	1.62	2.67
Other Turbine Trip (7)	--	--
Loss of Feedwater**(8)	4.62	--
Loss of Feedwater - Not Immediately Restorable**(8A)	--	6.74
Loss of Feedwater - Immediately Restorable*(8B)	--	--
Partial Loss of Feedwater*(8C)	--	--
Loss of Control Air (9)	7.26	2.47
Other Scram (10)	0.57	0.52
Main Steam Line Break - Outside Containment (11A)	--	--
Main Steam Line Break - Inside Containment (11B)	--	--
Feedwater Line Break-Outside Containment (12A)	--	--
Feedwater Line Break-Inside Containment (12B)	--	--
HPCI Steam Line Break (13)	--	--
RWCU Break-Return Line (14A)	--	--
RWCU Break-Suction Line (14B)	--	--
RCIC Steam Line Break (15)	--	--
Recirculation Discharge Line Break (16)	0.47	-
Recirculation Suction Line Break (17)	--	0.27
Core Spray Line Break-Inside Containment (18)	--	--

IMPORTANCE* OF INTERNAL INITIATING EVENTS

Internal Initiating Event Category (event no.)	January 1986 Version (%)	September 1987 Version (%)
Medium Steam Break I-Inside Containment (19A)	--	--
Medium Steam Break II-Inside Containment (19B)	--	--
Medium Water Break-Inside Containment (19C)	--	--
Small Steam Break-Inside Containment (20A)	0.66	0.77
Small Water Break-Inside Containment (20B)	5.05	7.03
Inadvertent Opening of 1-3 MSRVs (21A)	7.08	1.95
Inadvertent Opening of 4 or more MSRVs (21B)	--	--
Loss of Offsite Power Resulting in 1-3 Stuck Open Relief Valves	0.36	11.95
Transient Resulting in 1-3 Stuck Open Relief Valves	13.33	8.30
Transient Resulting in Loss of Feedwater	2.97	1.54
Transient Resulting in Feedwater Rampup	1.94	0.80
Transient Resulting in Small LOCA	--	--
Transient with subsequent Scram Failure	5.83	2.77
<hr/>		
Percent CMF Represented by Top 100 Sequences	70.87	76.29
Absolute CMF from Internal Events (per calendar year)	3.9×10^{-3}	4.7×10^{-4}

*Loss of Feedwater divided into three subcategories for the September 1987 version

NOTE: See section 5.3 for definition of Importance.

TABLE 6-1

PLANT DAMAGE STATE (PDS) FREQUENCIES

PDS	Frequency (Per Calendar Year)	
	January 1986	September 1986
	Version	Version
1A	6.91×10^{-4}	9.28×10^{-5}
1D	3.57×10^{-4}	3.72×10^{-5}
1J	2.31×10^{-6}	4.26×10^{-7}
1L	4.08×10^{-5}	8.27×10^{-6}
2A	4.65×10^{-5}	8.51×10^{-6}
2D	1.60×10^{-5}	2.46×10^{-6}
2L	2.12×10^{-6}	1.30×10^{-7}
3A	7.33×10^{-5}	1.24×10^{-7}
3D	1.98×10^{-5}	3.36×10^{-8}
5A	1.44×10^{-4}	2.17×10^{-6}
5D	4.05×10^{-5}	6.38×10^{-7}
6A	3.87×10^{-5}	1.17×10^{-7}
6D	1.05×10^{-5}	3.01×10^{-8}
6L	3.51×10^{-6}	4.22×10^{-11}
7A	5.12×10^{-4}	1.11×10^{-4}
7D	1.39×10^{-4}	3.90×10^{-5}
7J	1.48×10^{-6}	4.49×10^{-7}
8A	1.38×10^{-5}	4.71×10^{-6}
8D	3.76×10^{-6}	1.27×10^{-6}
9A	3.44×10^{-4}	3.10×10^{-5}
9D	9.60×10^{-5}	8.89×10^{-6}
9J	1.00×10^{-6}	1.27×10^{-7}
10A	1.02×10^{-4}	2.41×10^{-5}
10D	2.77×10^{-5}	6.53×10^{-6}
10L	3.23×10^{-5}	9.26×10^{-9}
13A	5.85×10^{-4}	7.16×10^{-5}
13D	2.81×10^{-4}	2.37×10^{-5}
13J	1.97×10^{-6}	2.79×10^{-7}
15A	2.31×10^{-4}	2.93×10^{-6}
15D	6.42×10^{-5}	8.16×10^{-7}
TOTAL	3.9×10^{-3}	4.7×10^{-4}

- NOTES: 1. Plant Damage States are defined in Table 6-1 Part B.
2. The values for the following Plant Damage States are $< 10^{-6}$ for both the January 1986 and the September 1987 versions:
 2B, 2C, 2F, 2J, 3J, 3L, 4A, 4B, 4C, 4D, 4F,
 4J, 4L, 5J, 5L, 6J, 7L, 8J, 8L, 9L, 10B, 10C,
 10F, 10J, 13L, 15J, 15L.

TABLE 6-1 Part B

DEFINITION OF PLANT DAMAGE STATES (PDS)

PDS (Numeric)	Core Melt Time < 6 hours after Event Initiation		Vessel Pressure At Melt < 400 psai		Debris Bed Cooling Availability		> 1 ft water in Drywell at Time of Melt	
	Yes	No	Yes	No	Yes	No	Yes	No
1	Yes		Yes			No		No
2	Yes		Yes			No	Yes	
3	Yes		Yes		Yes			No
4	Yes		Yes		Yes		Yes	
5	Yes			No		No		No
6	Yes			No		No	Yes	
7	Yes			No	Yes			No
8	Yes			No	Yes		Yes	
9		No	Yes			No		No
10		No	Yes			No	Yes	
11		No	Yes		**		**	**
12		No	Yes		**		**	**
13		No		No		No		No
14		No		No		No	Yes	
15		No		No	Yes			No
16		No		No	Yes		Yes	

PDS (Alpha)	Primary Containment Intact at time of Melt		Elevated Release*	Other Filtering Mechanisms		
	Yes	No		SP	SBGT	None
A	Yes	No	Yes	SP		
B	Yes				SBGT	
C	Yes					None
D	Yes			No	SP	
E	Yes			No		**
F	Yes			No		None
G		No	**	**	**	**
H		No	**	**	**	**
I		No	**	**	**	**
J		No		No	SP	
K		No		No		**
L		No		No		None

SP = Suppression Pool

SBGT = Standby Gas Treatment Charcoal Filter, No SP

*Includes SGBT Roughing and HEPA Filters

**No Possible (No SBGT) or Forbidden by the Model

TABLE 6-2

DOMINANT SEQUENCES

(Update of Table 2 of October 1, 1987, NRC Letter)

	<u>FREQUENCY</u>
1. Loss of Feedwater Transient, Failure of HPCI and RCIC, Failure of Manual ADS Blowdown, Electric Power State 16, CAS State 1, No Recovery.	2.25 E-5/Year
2. Small LOCA, Failure of Torus Cooling, Electric Power State 16, CAS State 1, No Recovery.	2.05 E-5/Year
3. Main Steam Isolation Valve Closure, Failure of HPCI and RCIC, Failure of Manual ADS Blowdown, Electric Power State 16, CAS State 1, No Recovery.	2.04 E-5/Year
4. Transient with 1-3 Stuck Open Relief Valves, Failure of Torus Cooling, Electric Power State 16, CAS State 1, No Recovery.	1.58 E-5/Year
5. Loss of Offsite Power, Failure of HPCI and RCIC, Failure of LPCI Injection, Failure of Torus Cooling, Failure of Core Spray, Electric Power State 1, CAS State 1, Recovery.	1.25 E-5/Year
6. Pressure Regulator Fails Closed, Failure of HPCI and RCIC, Failure of Manual ADE Blowdown. Electric Power State 16, CAS State 1, No Recovery.	1.09 E-5/Year
7. Loss of Condenser Vacuum, Failure of HPCI and RCIC, Failure of Manual ADS Blowdown, Electric Power State 16, CAS State 1, No Recovery.	9.91 E-6/Year
8. Loss of Offsite Power Resulting in 1-3 Stuck Open Relief Valves, Failure of EECW, Electric Power State 3, CAS State 1, Recovery	9.37 E-6/Year
9. Main Steam Isolation Valve Closure, Failure of HPCI and RCIC, Failure of the Condensate System, Failure of Manual ADS Blowdown, Electric Power State 16, CAS State 1, No Recovery.	8.67 E-6/Year
10. Transient with 1-3 Stuck Open Relief Valves, Failure of HPCI, Failure of RHR, Electric Power State 3, Recovery.	8.09 E-6/Year