UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

March 8 , 2007

MEMORANDUM TO: ACRS Members

FROM: Eric A. Thornsbury, ACRS Senior Staff Engineer

SUBJECT: CERTIFICATION OF THE MINUTES OF THE MEETING OF THE ACRS SUBCOMMITTEE ON RELIABILITY & PROBABILISTIC RISK ASSESSMENT, DECEMBER 14-15, 2006 - ROCKVILLE, MARYLAND

The minutes of the subject meeting, issued January 24, 2007, have been certified as the official record of the proceedings of that meeting. A copy of the certified minutes is attached.

Attachment: As stated

electronic cc: F. Gillespie
S. Duraiswamy
C. Santos

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

January 24, 2007

MEMORANDUM TO: George E. Apostolakis, Chairman
Reliability & Probabilistic Risk Assessment Subcommittee

FROM: Eric A. Thornsbury, ACRS Senior Staff Engineer

SUBJECT: WORKING COPY OF THE MINUTES OF THE MEETING OF THE ACRS SUBCOMMITTEE ON RELIABILITY & PROBABILISTIC RISK ASSESSMENT, DECEMBER 14-15, 2006 - ROCKVILLE, MARYLAND

A working copy of the minutes for the subject meeting is attached for your review. Please review and comment on them. If you are satisfied with these minutes, please sign, date, and return the attached certification letter.

Attachment: Minutes (DRAFT)

cc: Reliability & Probabilistic Risk Assessment Subcommittee Members
F. Gillespie
S. Duraiswamy
C. Santos

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

MEMORANDUM TO: Eric A. Thornsbury, ACRS Senior Staff Engineer

FROM: George E. Apostolakis, Chairman
Reliability & Probabilistic Risk Assessment Subcommittee

SUBJECT: CERTIFICATION OF THE MINUTES OF THE MEETING OF THE ACRS SUBCOMMITTEE ON RELIABILITY & PROBABILISTIC RISK ASSESSMENT, DECEMBER 14-15, 2006 - ROCKVILLE, MARYLAND

I do hereby certify that, to the best of my knowledge and belief, the minutes of the subject meeting on December 14-15, 2006, are an accurate record of the proceedings for that meeting.

George E. Apostolakis
Subcommittee Chairman

3/8/07
Date

Issued: January 24, 2007
Certified: March 8, 2007

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
MEETING OF THE ACRS SUBCOMMITTEE ON
RELIABILITY & PROBABILISTIC RISK ASSESSMENT
MEETING MINUTES - DECEMBER 14-15, 2006
ROCKVILLE, MARYLAND

INTRODUCTION

The ACRS Subcommittee on Reliability & Probabilistic Risk Assessment held a meeting on December 14-15, 2006, in Room T-2B1 and T-2B3, 11545 Rockville Pike, Rockville, MD. The purpose of this meeting was to discuss the probabilistic risk assessment (PRA) for the Economic Simplified Boiling Water Reactor (ESBWR), an advanced design from General Electric (GE) that is in the process of design certification. Eric Thornsbury was the Designated Federal Official for this meeting. The Committee received no written comments or requests for time to make oral statements from the public. The Subcommittee Chairman convened the meeting at 8:30 a.m. on December 14, 2006, recessed at 6:00 p.m., reconvened at 8:30 a.m. on December 15, 2006, and adjourned at 11:50 a.m..

ATTENDEES

ACRS

G. Apostolakis, Subcommittee Chairman
S. Abdel-Khalik, Member
M. Bonaca, Member
M. Corradini, Member
T. Kress, Member
O. Maynard, Member
W. Shack, Member
J. Sieber, Member
G. Wallis, Member
E. Thornsbury, Designated Federal Official

Principal Speakers

R. Wachowiak, GE
T. Kevern, NRC/NRO
L. Mrowca, NRC/NRO

Other members of the staff and public attended this meeting. A complete list of attendees is in the ACRS Office File and is available upon request. The presentation slides and handouts used during the meeting are attached to the office copy of these minutes.

OPENING REMARKS BY CHAIRMAN APOSTOLAKIS

George Apostolakis, Chairman of the ACRS Subcommittee on Reliability & Probabilistic Risk Assessment, convened the meeting at 8:30 a.m. Dr. Apostolakis stated that the purpose of this meeting was to review selected details of the ESBWR probabilistic risk assessment. He said the Subcommittee would gather information, analyze relevant issues and facts, and formulate proposed positions and actions, as appropriate, for deliberation by the full Committee. The potentially even use different types of squibs on the same valve. Mr. Maynard agreed that a testing program before installation of the squibs would provide a better chance of catching such errors.

- Dr. Kress asked about the construction material for the walkway over the BiMAC device, and noted that it may become part of the melt. Mr. Wachowiak agreed, and noted that they are considering the materials interactions due to the walkway and other sources such as the control rod drive mechanisms.

- Mr. Sieber commented that GE could develop many different instrumentation and control systems that meet the standards, which could range from bare-bones to a Cadillac. Mr. Wachowiak acknowledged the difficulties that posed for the PRA early in the design. He described how they began with a bare-bones approach in the PRA, but can now add more detail as they are actually designing the systems. Through their early modeling, he noted their identification of important common-mode failure concerns, and so suggested that a diverse I&C system be incorporated into the overall design.

- Dr. Apostolakis commented that the results of the uncertainty analysis, as presented, account for some of the uncertainties. He referred to the results of the sensitivity analyses, which led him to believe in a greater uncertainty, perhaps with a 95th percentile of CDF at 10^{-6}, which still is a good design. Mr. Wachowiak acknowledged his concerns, and added that the use of the number is also an important consideration in how they calculate it. He pointed out that even when qualitatively assessing other uncertainties not in the calculation, most of the risk curve for the ESBWR is will within the Commission's goals. Dr. Kress suggested that a better approach would assume that we constructed the goals with this in mind, and that the only the calculated values must meet the goals.

<u>Dominant Accident Sequences</u>

Mr. Wachowiak next described the dominant accident sequences from Revision 1 of the ESBWR PRA, which the subcommittee had requested to see during the previous meeting. Mr. Wachowiak provided event trees for loss-of-feedwater events and general transients, with which he described the response of the plant which can lead to the most likely accident sequences.

<u>Comments and Observations From the Subcommittee Members</u>

- Dr. Abdel-Khalik asked Mr. Wachowiak to clarify the plant's response to a LOCA. Mr. Wachowiak described the response, and noted that no design-basis accidents lead to the reactor water level reaching the low level signal where the equalizing line is called to open.

- Mr. Maynard asked about the source of reliability data for diesel generators and other electrical equipment which is not safety-related in this plant design. Mr. Wachowiak replied that they are using data from existing plants, but also pointed out that though their maintenance may be different, their performance requirements are also less stringent. Mr. Maynard stated that he believes that some regulatory treatment of such non-safety systems may be necessary to ensure consistency with the PRA.

- Dr. Apostolakis raised the issue of dependence among human errors, specifically noticing two related human errors in the dominant sequences, the mispositioning of two control rod drive (CRD) system valves. Mr. Wachowiak agreed with the importance of examining such dependencies, and described how they modified the plant design to reduce the importance of the CRD system. He also discussed how GE is considering instrumentation on such important valves to alert operators of their position.

- Dr. Apostolakis commented on the possible failures of passive systems, and asked if GE was concerned about the thermal-hydraulic uncertainties that could affect the operation of the passive systems. Mr. Wachowiak replied that they are in ongoing dialogue with the staff. He stated that their thermal-hydraulic calculations show that very, very small parameter values are necessary in order to lead to problems, and so their current success criteria are conservative.

- Dr. Shack asked about the contribution of spurious relief valve operation to the LOCA frequencies. Mr. Wachowiak described how two types of spurious actuations were considered, the spurious opening of one or more valves, and a spurious initiation signal for the valves. Either way, the sequence created is a steam LOCA, which is easy to handle in the ESBWR and is not a dominant contributor to the risk.

- Dr. Abdel-Khalik followed up with a question regarding the contribution of steam-line breaks outside containment that could create a bypass. Mr. Wachowiak answered that they are also low contributors due to their low initiation frequency and need for isolation failure before they are a concern.

PRA Update and Information Exchange

Mr. Wachowiak's next presentation discussed some of the changes to the plant design which were going to be incorporated into the next revision of the PRA and the effects of those changes on the results. He first listed and discussed some of the changes to the base PRA model, which included additional water volume in the isolation condenser system, more detailed design of the I&C architecture, additional design detail for balance-of-plant systems, a revised common cause model, additional model detail for medium-time sequences, and other minor changes to make the PRA more usable.

The first detailed discussion of changes by Mr. Wachowiak regarded the changes to the isolation condenser system. By adding nine cubic meters of additional water volume in each isolation condenser, the new design optimizes the emergency core cooling system actuation settings, which reduces the top 90% of cutsets in the Revision 1 PRA. Mr. Wachowiak illustrated the design changes with several stand-alone figures describing the details of the system.

Mr. Wachowiak then explained the development of design details for the architecture of the digital instrumentation and control system. He described that GE has determined the diversity and defense-in-depth requirements for the I&C systems, and that they have chosen to implement a double-failure-proof safety related digital I&C system. Mr. Wachowiak described how such a design still allows single-failure protection with one division out of service, which enables on-line battery testing. He illustrated the concept with the example arrangement for squib valve actuation, then explained the I&C system design in more detail with the use of several block diagrams.

Mr. Wachowiak completed this session by describing the effects of the changes in Revision 2, and estimated that the Revision 2, Level 1 internal events model results would be available in April 2007. They plan to extrapolate those results to estimate the effects on Level 2 and external events to also produce a revised DCD Chapter 19 at the same time. By September, they need to have the complete Revision 2 of the PRA to support combined license applications.

<u>Comments and Observations From the Subcommittee Members</u>

- Dr. Apostolakis asked how the I&C in included in the PRA. Mr. Wachowiak described first how they captured the hardware configuration, then stated that they treat the internal software operation as a simple common cause failure. He noted that the I&C systems for the ESBWR do not have much control, but are mostly just comparative threshold trip systems. They use a common cause software failure probability of 10^{-5}, based on the reliability of commercial software systems. He acknowledged that this is only an assumed number at this time. He also noted that GE has decided to implement a diverse system for many functions due to the uncertainties. Dr. Apostolakis recommended staying away from the numbers, but to base the argument on qualitative arguments based on functionality and diversity.

- Dr. Wallis asked if the PRA runs a thermal-hydraulic code in order to know what the water level in the reactor vessel is. Mr. Wachowiak replied that it does not directly run a thermal-hydraulic code, but knows the water level based on the initiating event, which they calculate in separate thermal-hydraulic codes. For example, they know that the water level drops below Level 2 in loss-of-feedwater events.

- Dr. Abdel-Khalik asked if these supporting thermal-hydraulic calculations are best estimates. Mr. Wachowiak replied affirmatively, but noted that due to the evolving design, they may not be the best best-estimate.

- Dr. Wallis questioned the use of qualitative arguments in the PRA to dismiss some failures such as reactor water level instrumentation failure. Mr. Wachowiak acknowledged the potential importance of these issues and noted that as more design details become available, they are being incorporated directly into the PRA.

- Dr. Apostolakis asked how they addressed the issue of common cause failure in the I&C system. Mr. Wachowiak explained how they use four separate and independent trains. Within systems, the trains are separated, but common cause failure could still occur due to the same manufacturer. Different systems, however, come from different manufacturers. Therefore, there is redundancy within systems and diversity among systems. Mr. Maynard agreed with this approach.

- Dr. Wallis questioned where some of the failure probabilities come from, such as for the load drivers. Dr. Apostolakis stated that many of the numbers come from light water reactor experience. Mr. Wachowiak stated that one goal of the PRA is use sensitivity analyses to show that the goals are met regardless of what data sets they use.

However, he stated that when they purchase components such as load driver cards, they will ask for failure data and update the PRA accordingly.

- Dr. Apostolakis commented on the sensitivity analyses and the inherent uncertainty in the calculated risk values. He noted that we must wonder if things that are left out of the PRA are more likely than the calculated risk. He reminded everyone that, in the end, we grant a license based on the fact that the design meets all the regulatory requirements, both deterministic and probabilistic, and suggested that we should not be too hung up on the numbers.

- Dr. Corradini asked about the low risk results for the external events, which surprised him. Mr. Wachowiak explained that the external events include internal fires and internal floods which are not site-dependent. For site-dependent external events, such as external floods, GE has specified that plants must meet certain siting parameters, which help to keep the risk low.

- Dr. Abdel-Khalik asked whether the detonation of multiple squib charges on a valve can be a problem. Mr. Wachowiak replied that it does not. Dr. Abdel-Khalik also asked if the detonations could cause a pipe failure. Mr. Wachowiak replied that the design requirements for the piping take that into account.

PRA Modeling Issues

On the second day of the meeting, Mr. Wachowiak began with a presentation of several PRA modeling issues raised by the subcommittee in response to the previous meeting. He discussed the common cause failure methods, the treatment of failure rates for components with long test intervals, and the treatment of thermal-hydraulic uncertainties.

Mr. Wachowiak first described the use of the Alpha Factor common cause method in Revision 1 of the PRA, and the problems it caused regarding uncertainty analysis and sensitivity analyses. He stated that Revision 2 will utilize the Multiple Greek Letter (MGL) Method, which the CAFTA PRA software supports. Mr. Wachowiak demonstrated the use of the MGL method through screenshots from the software.

Mr. Wachowiak continued by describing the approach used to estimate failure rates for equipment with longer test intervals than represented by the available data. He noted that most demand-failure data is associated with quarterly-tested equipment, while the ESBWR will have equipment that licensees may test less frequently. He described the three methods they considered and noted that they only used two, since no components fit into the second approach. The first approach applies directly for equipment with a test interval of six months or less. The third approach applies to equipment with test intervals greater than one year. For these, they convert the demand failure to a rate, then calculate the unavailability based on this rate, the proposed test interval, and the assumption of no repair.

For the last part of this session, Mr. Wachowiak discussed GE's approach to thermal-hydraulic uncertainty for the PRA. He explained that the PRA success criteria are considered to be bounding, with very few cases involving uncovery of the fuel. In those few cases, they do not calculate any significant fuel heatup. He briefly described their original plan for addressing thermal-hydraulic uncertainty, then described their current plan. It should minimize their reliance on complex thermal-hydraulic calculations by using sensitivity analyses to quantify the PRA with the even-more-conservative design-basis success criteria for the passive safety systems. They expect to be able to show a very small effect on the core damage frequency and large release frequency, thereby demonstrating that the thermal-hydraulic uncertainties will have little or no effect on the PRA. They would provide additional thermal-hydraulic calculations only for outlying sequences. This approach will utilize Revision 2 of the PRA, and Mr. Wachowiak stated that they expect to have a report in May.

<u>Comments and Observations From the Subcommittee Members</u>

- Dr. Apostolakis questioned what the designers are doing to try to reduce the common cause failure factors. Mr. Wachowiak commented that designers tend to think that a robust design eliminates common cause failures, but that they are taking into account other issues such as the operating environment. Dr. Wallis and Dr. Corradini commented that design errors could increase the common cause factor. Mr. Wachowiak reiterated his earlier description of the process used to reduce the common cause failure of squib valves through the use of different types of squib valves from different manufacturers.

- Dr. Apostolakis asked why the PRA is using data values from the Utility Requirements Document. Dr. Shack suggested that it was a good idea, since it highlights the differences between designs without relying on differences in data. Dr. Kress agreed. Dr. Apostolakis countered that our job is not to compare designs, but to compare to the Commission's goals, which argues for using the best available data.

- Dr. Wallis asked about the uncertainty in the thermal-hydraulic calculations, such as the reactor vessel water level. Mr. Wachowiak replied that they would discuss this in the report being prepared for the staff. He explained how GE has used the MAAP code for their PRA-related thermal-hydraulic calculations, while using TRACG for their design-basis calculations. Dr. Corradini stated his hope to see benchmark calculations between TRACG and MAAP to compare the codes. Mr. Wachowiak explained that such comparisons exist for some scenarios, but not for the full range included in the PRA, so that they believe they can sufficiently trust the MAAP predictions.

- Dr. Abdel-Khalik asked what we can learn from emergency operations at current plants related to core uncovery. Mr. Wachowiak replied that we know from the current plants that if the core is uncovered, it will not suffer damage if reflooded in a fairly short period of time.

- Dr. Shack suggested that GE present some parametric input uncertainty calculations and identify the effects of those uncertainties. Dr. Corradini agreed with the suggestion. However, Mr. Sieber pointed out the distinction between the thermal-hydraulic question and the PRA question. Dr. Wallis suggested the need for a separate thermal-hydraulics meeting to explore the issue further. Several Members and Mr. Wachowiak discussed the different needs of the thermal-hydraulic analyses and the PRA analysis, and Dr. Apostolakis concluded that we should discuss the issue of the effects of thermal-hydraulic uncertainties on the PRA at the next PRA subcommittee meeting, though he also indicated that some calculations showing the effects of input uncertainties would be helpful.

Regulatory Treatment of Non-Safety Systems

Due to time constraints, Mr. Wachowiak and the Members decided to skip the slide package discussing the summary of the external events PRA. Mr. Wachowiak then proceeded with a discussion of GE's strategy for the regulatory treatment of non-safety systems (RTNSS) for the ESBWR standard design. He first described the sources of the requirements for RTNSS, which include several Commission papers, associated staff requirements memoranda, and the precedents set by previous design certifications. He noted that there are both deterministic and probabilistic selection criteria, and that they determined potential RTNSS systems based on the need to provide functions to address ATWS, station blackout, long-term cooling, seismic events, adverse systems interactions, and the probabilistic safety goals.

Mr. Wachowiak then described their assessment of each of the required functions and the resulting RTNSS identifications. For ATWS mitigation, they identified the alternate rod injection (ARI) system and the feedwater controller for potential RTNSS. Neither station blackout nor seismic considerations introduce any additional RTNSS concerns. For long-term safety, Mr. Wachowiak described four functions that they must maintain: core cooling, decay heat removal, post-accident monitoring, and control room habitability. The design of the ESBWR provides for 72 hours of cooling without operator intervention, then provides on-site resources to maintain safety from that point until seven days, then requires commodity replacement (such as water or fuel) from offsite after seven days. To meet the probabilistic safety goals, some systems may require simple treatment through technical specifications. A subset of this requirement, systems needed to address uncertainty in the probabilistic estimates, produces no new RTNSS. Mr. Wachowiak also discussed their evaluation of RTNSS based on initiating events and adverse systems interactions. Mr. Wachowiak completed his presentation by listing the proposed systems for RTNSS and discussed the proposed treatment requirements for some of the selected systems.

Comments and Observations From the Subcommittee Members

- Dr. Apostolakis asked about the connection between the RTNSS process and the categorization of 50.69 that identifies risk-significant, non-safety systems. Mr. Wachowiak replied that we cannot connect the two issues because the Commission provided the instructions for RTNSS separately through the different mechanisms discussed on the slide. He acknowledged the similarities, but noted that many risk-related programs have different requirements, and that the RTNSS is different from 50.69, which are both different from the maintenance rule, and different from the D-RAP guidance.

- Dr. Shack asked what details the PRA identified that will go back into the design control document. Mr. Wachowiak cited the example of the configuration of the instrumentation and control system, and noted that they are working to compile a complete list of items that went into design requirements that came out of the PRA.

- Dr. Wallis asked which RTNSS systems help meet the CDF criterion. Mr. Wachowiak replied that none do, but that the large release frequency is more challenging. He explained that a catastrophic failure of the digital instrumentation and control system can lead to loss of emergency core cooling and containment isolation. Therefore, they identified parts of the diverse protection system as RTNSS to meet the large release criterion.

- Dr. Apostolakis questioned the need to identify RTNSS to address uncertainty. Mr. Wachowiak explained that as a process to identify systems needed to make sure that the uncertainty band is below the probabilistic goals. Dr. Apostolakis stated that the goals are a mean value, and that nobody says that the 95th percentile needs to meet the goal, though it is OK to attempt to address it. He believes it to be a bad precedent. Mr. Wachowiak stated that their goal is to design a plant that is much safer that required, and, while he shares Dr. Apostolakis' views, he stated that precedent does not do it that way.

ESBWR PRA Staff Update

Mr. Tom Kevern, NRO, began the staff's presentation. He first discussed the status of the staff's requests for additional information (RAIs) related to the ESBWR PRA, stating that the staff has received responses to 84 of the 157 RAIs issued to date, with 15 of those requiring supplemental responses. Mr. Kevern also discussed the schedule for the review of the PRA in coordination with the design control document (DCD) and the staff's approach to parallel reviews of the DCD, PRA, and soon-to-be-submitted combined license applications.

Ms. Lynn Mrowca, NRO, then continued the staff's presentation by listing and describing the key technical issues the staff has identified in their RAIs. In the Level 1, full-power PRA, she described the staff's questions regarding common cause failure probabilities, the modeling of instrumentation and control systems, the mission time of the PRA, the modeling of steam suppression vacuum breakers, fire risk issues related to spurious valve actuation, input from the PRA to the licensing basis, and thermal-hydraulic success criteria and uncertainty. For the shutdown PRA, she and Ms. Maria Pohida discussed their questions related to the early aspect of the large release frequency, the role of the operator in inducing accidents or manually operating equipment, common cause failures of non-safety systems, and the impact of an open containment on the PRA. Ms. Mrowca completed the staff's presentation with a brief discussion of the staff's RAIs related to the Level 2 PRA, including questions on the BiMAC system and the operation of vacuum breakers.

<u>Comments and Observations From the Subcommittee Members</u>

- Mr. Maynard asked how much of the requests for additional information are required. Mr. Kevern replied that the staff's position is that they are all required, but that in some cases, the wording of the question asks for a clarification so that the staff and applicant have a common understanding.

- Dr. Apostolakis asked for the staff's opinion regarding the use of different data sources for the PRA. Mr. Nick Saltos, NRR, answered that the applicant is not required to go to the Utility Requirements Document, but for some components, they do not have any other source. In general, the staff wants them to use the best available data sources, though more recent does not necessarily mean more reliable.

- Dr. Apostolakis asked how the staff handles something for which there are no accepted models for calculating failure, such as digital systems. Mr. Saltos replied that they will certify the design with the state-of-the-art, but use a conservative approach where necessary. For example, they will look at high level attributes such as separation, number of divisions, redundancy, and similar features. Dr. Apostolakis asked if they would put numbers on the performance of I&C systems. Mr. Saltos replied that they would be comfortable making a decision for RTNSS based on a failure rate of 10^{-3}, since that seems supportable based on data from other industries, and given that the system meets the regulatory requirements for defense-in-depth and diversity. Dr. Apostolakis replied that they should rely on those regulatory requirements, as there is no basis for any number. He stated that defense-in-depth is the key.

- Dr. Shack asked what the staff expects in regard to thermal-hydraulic uncertainty. Mr. Saltos explained how they addressed the issue for the AP600 and AP1000, which he understands is similar to what GE will do. Westinghouse did not calculate the uncertainties, but bounded them and demonstrated that the system could do the job with the assumed success criteria. They first identified the low-thermal-hydraulic-margin, risk-significant scenarios, then performed calculations for those scenarios using their design-basis code. So the staff would like to see similar calculations for the ESBWR.

- Dr. Abdel-Khalik asked about the possibility of noncondensable gases in the injection lines due to an error in startup procedures. Mr. Ralph Landry, NRR, stated that they have not looked at that exact problem, but that they were doing an audit this week that included a number of questions on noncondensable gas transfer.

- Dr. Apostolakis asked if the staff applies the same scrutiny to operator models as to thermal-hydraulic codes, given that the PRA uses a human reliability analysis model that the staff has not reviewed. Mr. Saltos stated that they do have some RAIs related to human reliability, but that they have the overall impression that the numbers are conservative. Dr. Apostolakis agreed that the numbers are reasonable, but noted that we may see new failure modes due to different operator response times.

Closing Discussions

Closing Comments and Observations From the Subcommittee Members

- Mr. Sieber noted that he struggled through some of his review parts because of a lack of sufficient information in the design control document. He expects to be able to do a better job once he sees the new revisions. He cautioned that there is a lot of work ahead in order to come to a conclusion regarding the acceptability of the PRA.

- Mr. Maynard also stated that both the staff and the Committee have quite a bit to do to reach a success path. He stated that there seems to be a lot of uncertainty regarding what it will take to satisfy the Committee and the staff.

- Dr. Kress stated that PRA looks pretty good and is very comprehensive, though he is anxious to see the uncertainty analysis. One of his key issues is the thermal-hydraulic uncertainty, and he likes GE's approach. He suggested that the staff do some of their own benchmarking, perhaps with RELAP.

- Dr. Abdel-Khalik noted his biggest concern is with the common cause failure of squib valves, such as due to a supplier providing the wrong squibs. He acknowledged that testing procedures may help, but that they may need to include such a possibility in the estimate of the failure probability. He also reiterated his concern regarding trapped gases in the lines.

- Dr. Bonaca stated that the design of the plant impressed him, and it indicates that we have learned from the current generation of plants. They seem to have taken every opportunity to ensure a robust design. He also stated that the PRA impressed him, though clearly they still need to assemble some pieces.

- Dr. Apostolakis emphasized that the extensive discussions on many issues should not cloud the fact that it is a very good PRA. He noted that we still have some work to do regarding thermal-hydraulic uncertainties, but that he did not see any showstoppers.

- Dr. Corradini stated agreement with Mr. Maynard's concern that there may not be a clear path forward for determining the acceptability of the PRA.

SUBCOMMITTEE DECISIONS AND ACTIONS

The subcommittee raised several issues to discuss at future meetings, and decided that no interim letter was necessary at this time. The next meeting will focus on thermal-hydraulic uncertainty, the Level 2 PRA, and severe accident phenomena.

BACKGROUND MATERIALS PROVIDED TO THE SUBCOMMITTEE PRIOR TO THIS MEETING

Documents	
1.	S. Traiforos, et al., "Letter Report: Preliminary Review of the Economic Simplified Boiling Water Reactor (ESBWR) Probabilistic Risk Assessment (PRA)," Link Technologies, November 13, 2006.

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Note: Additional details of this meeting can be obtained from a transcript of this meeting available for downloading or viewing on the Internet at http://www.nrc.gov/what-we-do/regulatory/advisory/acrs.html or purchase from Neal R. Gross and Co., Inc., (Court Reporters and Transcribers) 1323 Rhode Island Avenue, NW, Washington, DC 20005 (202) 234-4433.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

November 28, 2006

MEMORANDUM TO: Michael R. Snodderly, Branch Chief, ACRS/ACNW

FROM: Eric A. Thornsbury, Senior Staff Engineer

SUBJECT: FEDERAL REGISTER NOTICE REGARDING THE MEETING OF THE ACRS SUBCOMMITTEE ON RELIABILITY AND PROBABILISTIC RISK ASSESSMENT, DECEMBER 14 AND 15, 2006, ROCKVILLE, MARYLAND

Attached is a Federal Register Notice regarding the subject meeting. Please have this Notice transmitted for publication as soon as possible.

Attachment:
FR Notice

cc with Attachment:
G. Apostolakis, ACRS
J. Larkins, ACRS
B. Sosa, OEDO
J. Szabo, OGC
A. Bates, SECY
S. Burnell, OPA
M. Weber, NRR
G. Holahan, NRO
D. Matthews, NRO
A. Cubbage, NRO
T. Kevern, NRO
M. Shuaibi, NRO
PMNS
Public Document Room

[7590-01-P]

NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS)
MEETING OF THE ACRS SUBCOMMITTEE ON RELIABILITY
AND PROBABILISTIC RISK ASSESSMENT

Notice of Meeting

The ACRS Subcommittee on Reliability and Probabilistic Risk Assessment (PRA) will hold a meeting on December 14 and 15, 2006, Room T-2B1, 11545 Rockville Pike, Rockville, Maryland.

The entire meeting will be open to public attendance.

The agenda for the subject meeting shall be as follows:

Thursday, December 14, 2006 - 8:30 a.m. until the conclusion of business

Friday, December 15, 2006 - 8:30 a.m. until the conclusion of business

The Subcommittee will review the PRA for General Electric's next generation simplified boiling water reactor, the ESBWR. The Subcommittee will hear presentations by and hold discussions with representatives of the NRC staff and industry regarding this matter. The Subcommittee will gather information, analyze relevant issues and facts, and formulate proposed positions and actions, as appropriate, for deliberation by the full Committee.

Members of the public desiring to provide oral statements and/or written comments should notify the Designated Federal Official, Mr. Eric A. Thornsbury, (Telephone: 301-415-8716) five days prior to the meeting, if possible, so that appropriate arrangements can be made. Electronic recordings will be permitted.

Further information regarding this meeting can be obtained by contacting the Designated Federal Official between 7:30 a.m. and 4:15 p.m.(ET). Persons planning to attend this meeting are urged to contact the above named individual at least two working days prior to the meeting to be advised of any potential changes to the agenda.

11-28-2006
Date

Michael R. Snodderly
Michael R. Snodderly, Branch Chief, ACRS/ACNW

Advisory Committee on Reactor Safeguards
Reliability & Probabilistic Risk Assessment Subcommittee Meeting
Rockville, MD
14-15 December 2006

- Proposed Agenda -
Revision 12/01/06
Cognizant Staff Engineer: Eric Thornsbury (301-415-8716, eat2@nrc.gov)

	Topic	Presenter(s)	Time
December 14			
	Opening Remarks and Objectives	G. Apostolakis, ACRS	8:30 - 8:40 am
I	Overview of ESBWR PRA	R. Wachowiak, GE	8:40 - 10:00 am
	Break		10:00 - 10:15 am
II	Significant Changes in Revisions 1 and 2	R. Wachowiak, GE	10:15 am - 12:00 noon
	Lunch		12:00 noon - 3:00 pm
III	Discussion of Issues Raised During Previous ACRS Meeting	R. Wachowiak, GE	3:00 - 6:00 pm
	Recess for the day		6:00 pm

	Topic	Presenter(s)	Time
December 15			
	Reconvene		8:30 am
III	Completion of ESBWR Presentations	R. Wachowiak, GE	8:30 - 10:15 am
	Break		10:15 - 10:30 am
IV	Key Issues in Staff RAIs	T. Kevern, NRR/NRO N. Saltos, NRR/NRO M. Pohida, NRR/NRO R. Palla, NRR/NRO	10:30 - 11:30 am
	Closing Discussions	G. Apostolakis, ACRS	11:30 am - 12:00 noon
	Adjourn		12:00 pm

Notes:
- Presentation time should not exceed 50% of the total time allocated for a specific item.
- Number of copies of presentation materials to be provided to the ACRS - 35.

Tom Kevern
Lynn Mrowca

Hossein Hamzeehee
cellunoid
Solenoid

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

SUBCOMMITTEE MEETING ON RELIABILITY AND

PROBABILISTIC RISK ASSESSMENT

December 14, 2006
Date

NRC STAFF SIGN IN FOR ACRS MEETING

PLEASE PRINT

	NAME	NRC ORGANIZATION
1	Mohammed Shuaibi	NRC/NRO/DNRL
2	Amy Cubbage	NRO/NGE1
3	Tim Kress	NRO/NGE1
4	Lynn Mrowca	NRR/DRA
5	Don Dube	NRR/DRA
6	Ed Fuller	NRR/DRA
7	Mark Caruso	NRR/DRA
8	Theresa Clark	NRR/DRA
9	Marie Pohida	NRR/DRA
10	Nick Saltos	NRR/DRA
11	Malcolm Patterson	NRR/DRA
12	RALPH LANDRY	NRR/DSS
13	Walton Jensen	NRR/DSS
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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

SUBCOMMITTEE MEETING ON RELIABILITY AND

PROBABILISTIC RISK ASSESSMENT

December 15, 2006
Date

NRC STAFF SIGN IN FOR ACRS MEETING

PLEASE PRINT

	NAME	NRC ORGANIZATION
1	Hossein Hamzehee	NRR/DE
2	DON DUBE	NRR/DRA
3	Mark Caruso	NRR/DRA
4	Nick Saltos	NRR/DRA
5	M. GAVRILAS	RES/DRASP
6	MARIE POHIDA	NRR/DRA
7	MALCOLM PATTERSON	NRR/DRA
8	Theresa Clark	NRR/DRA
9	Charles Ader	NRO/DSRA
10	[illegible]	NRR/DRA
11	Rajendrasinh Solanki	RES/DRASP/PRAB
12	Amy Cubbage	NRO/DNRL/NGE1
13	Lynn Mrowca	NRR/DRA
14	BOB PALLA	NRR/DRA
15	Hanh Phan	NRR/DRA
16	Ed Fuller	NRR/DRA
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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

SUBCOMMITTEE MEETINGS ON RELIABILITY AND
PROBABILISTIC RISK ASSESSMENT

December 14, 2006
Date

PLEASE PRINT

	NAME	AFFILIATION
1	Rick WACHOWIAK	GE
2	LOU LANESE	GE / PANLYON TECHNOLOGIES
3	MIKE JONZEN	AREVA
4	Yumi kawanago	Mitubishi
5	CJ FONG	NRC RII / MIT
6	Patricia Campbell	GE
7	Hanh Phan	NRR
8	Ranjitsinh Parmar	NRR/DRA
9	~~Tim Kevern~~	
10	J. Alan Beard	GE Nuclear
11	Jim Fulford	ISL
12	Rajendrasinh solanki	RES/DRASP/PRAB
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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

SUBCOMMITTEE MEETINGS ON RELIABILITY AND
PROBABILISTIC RISK ASSESSMENT

December 15, 2006
Date

PLEASE PRINT

	NAME	AFFILIATION
1	MIKE JONZEN	AREVA
2	JIM FULFORD	ILS
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eAT

4. Procedure for Submitting Requests to Participate in Roundtable Discussions and for Submitting Written Comments.

Requests to Participate in Roundtable Discussions. The roundtable discussions will be open to the public. Persons wishing to participate in the discussions must submit a written request to the Section 108 Study Group. The request to participate must include the following information: (1) the name of the person desiring to participate; (2) the organization(s) represented by that person, if any; (3) contact information (address, telephone, telefax, and e–mail); and (4) a written summary of *no more than four pages* identifying, in order of preference, in which of the three general roundtable topic areas the participant (or his or her organization) would most like to participate and the specific questions the participant wishes to address in each topic area.

Space and time constraints may require that participation be limited in one or more of the topic areas, and it is likely that not all requests to participate can be accommodated. Identification of the desired topic areas in order of preference will help the Study Group to ensure that participants will be heard in the area(s) of interest most critical to them. The Study Group will notify each participant in advance of his or her designated topic area(s).

Note also for those who wish to attend but not participate in the roundtables that space is limited. Seats will be available on a first–come, first–served basis. All discussions will be transcribed, and transcripts subsequently made available on the Section 108 Study Group Web site (*http://www.loc.gov/section108*).

Written Comments. Written comments must include the following information: (1) the name of the person making the submission; (2) the organization(s) represented by that person, if any; (3) contact information (address, telephone, telefax, and e–mail); and (4) a statement of no more than 10 pages, responding to any of the topic areas or specific questions in this notice.

Submission of Both Requests to Participate in Roundtable Discussions and Written Comments. In the case of submitting a request to participate in the roundtable discussions or of submitting written comments, submission should be made to the Section 108 Study Group by e–mail (preferred) or by hand delivery by a commercial courier or by a private party to the address listed above. Submission by overnight delivery service or regular mail will not be effective due to delays in processing receipt.

If by e–mail (preferred): Send to the e–mail address *section108@loc.gov* a message containing the information required above for the request to participate or the written submission, as applicable. The summary of issues (for the request to participate in the roundtable discussion) or statement (for the written comments), as applicable, may be included in the text of the message, or may be sent as an attachment. If sent as an attachment, the summary of issues or written statement must be in a single file in either: (1) Adobe Portable Document File (PDF) format, (2) Microsoft Word version 2000 or earlier, (3) WordPerfect version 9.0 or earlier, (4) Rich Text File (RTF) format, or (5) ASCII text file format.

If by hand delivery by a private party or a commercial, non–government courier or messenger: Deliver to the address listed above a cover letter with the information required, and include two copies of the summary of issues or written statement, as applicable, each on a write–protected 3.5–inch diskette or CD–ROM, labeled with the legal name of the person making the submission and, if applicable, his or her title and organization. The document itself must be in a single file in either (1) Adobe Portable Document File (PDF) format, (2) Microsoft Word Version 2000 or earlier, (3) WordPerfect Version 9 or earlier, (4) Rich Text File (RTF) format, or (5) ASCII text file format.

Anyone who is unable to submit a comment or request to participate in electronic form (either through e–mail or hand delivery of a diskette or CD–ROM) should submit, with a cover letter containing the information required above, an original and three paper copies of the summary of issues (for the request to participate in the roundtable discussions) or statement (for the written comments) by hand to the appropriate address listed above.

Dated: November 28, 2006

Marybeth Peters,

Register of Copyrights.

[FR Doc. E6–20480 Filed 12–1–06; 8:45 am]

BILLING CODE 1410–21–F

NATIONAL TRANSPORTATION SAFETY BOARD

SES Performance Review Board

AGENCY: National Transportation Safety Board.

ACTION: Notice.

SUMMARY: Notice is hereby given of the appointment of members of the National Transportation Safety Board Performance Review Board.

FOR FURTHER INFORMATION CONTACT: Anh Bolles, Chief, Human Resources Division, Office of Administration, National Transportation Safety Board, 490 L'Enfant Plaza, SW., Washington, DC 20594–0001, (202) 314–6355.

SUPPLEMENTARY INFORMATION: Section 4314(c)(1) through (5) of Title 5, United States Code requires each agency to establish, in accordance with regulations prescribed by the Office of Personnel Management, one or more SES Performance Review Boards. The board reviews and evaluates the initial appraisal of a senior executive's performance by the supervisor, and considers recommendations to the appointing authority regarding the performance of the senior executive.

The following have been designated as members of the Performance Review Board of the National Transportation Safety Board. This list published previously on Friday, November 24, 2006. However, a change to membership has occurred since that time and here is the updated membership list.

The Honorable Robert L. Sumwalt, Vice Chairman, National Transportation Safety Board; PRB Chair.
The Honorable Deborah A.P.hersman, Member, National Transportation Safety Board.
Steven Goldberg, Chief Financial Officer, National Transportation Safety Board.
Lowell Martin, Deputy Executive Director, Consumer Products Safety Commission.
Frank Battle, Deputy Director of Administration, National Labor Relations Board.
Joseph G. Osterman,Managing Director, National Transportation Safety Board.

Dated: November 29, 2006

Vicky D'Onofrio,

Federal Register Coordinator.

[FR Doc. 06–9502 Filed 12–1–06; 8:45 am]

BILLING CODE 7533–01–M

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards (ACRS)

Meeting of the Acrs Subcommittee on Reliability and Probabilistic Risk Assessment; Notice of Meeting

The ACRS Subcommittee on Reliability and Probabilistic Risk Assessment (PRA) will hold a meeting on December 14 and 15, 2006, Room T–2B1, 11545 Rockville Pike, Rockville, Maryland.

The entire meeting will be open to public attendance.

The agenda for the subject meeting shall be as follows:

Thursday, December 14, 2006–8:30 a.m. until the conclusion of business.

Friday, December 15, 2006–8:30 a.m. until the conclusion of business.

The Subcommittee will review the PRA for General Electric's next generation simplified boiling water reactor, the ESBWR. The Subcommittee will hear presentations by and hold discussions with representatives of the NRC staff and industry regarding this matter. The Subcommittee will gather information, analyze relevant issues and facts, and formulate proposed positions and actions, as appropriate, for deliberation by the full Committee.

Members of the public desiring to provide oral statements and/or written comments should notify the Designated Federal Official, Mr. Eric A. Thornsbury, (Telephone: 301–415–8716) five days prior to the meeting, if possible, so that appropriate arrangements can be made. Electronic recordings will be permitted.

Further information regarding this meeting can be obtained by contacting the Designated Federal Official between 7:30 a.m. and 4:15 p.m. (ET). Persons planning to attend this meeting are urged to contact the above named individual at least two working days prior to the meeting to be advised of any potential changes to the agenda.

Dated: November 28, 2006.

Michael R. Snodderly,

Branch Chief, ACRS/ACNW.

[FR Doc. E6–20411 Filed 12–1–06; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Maximum 40-Year Licensing Terms for Certain Fuel Cycle Facilities

AGENCY: Nuclear Regulatory Commission.

ACTION: Notice.

SUMMARY: The U.S. Nuclear Regulatory Commission (NRC) has established a new policy extending the maximum license term for certain 10 CFR Part 70 fuel cycle licensees who are required to submit Integrated Safety Analysis (ISA) summaries for approval. Such license terms are being extended from the current 10-year period to a 40-year period, on the next renewal of the affected license. The NRC is also extending the maximum license term to a 40-year period for new 10 CFR Part 70 license applicants, where the applicant is required to submit an ISA summary for approval. The 10-year term has been a matter of policy and practice since 1990 (55 FR 24948; June 19, 1990); it is not codified in the regulations.

The NRC added Subpart H requirements to 10 CFR part 70 on September 18, 2000 (65 FR56211). The Subpart H requirements apply to licensees possessing greater than a critical mass of special nuclear material. Under Subpart H, both new applicants and existing licensees are required to conduct an ISA and submit an ISA summary to the NRC for approval. An ISA is a systematic analysis to identify facility and external hazards; potential accident sequences, including likelihood and consequences; and items relied on for safety to prevent potential accidents or mitigate the consequences.

Licensees are required to keep their ISAs up-to-date. In addition to the initial ISA summary, licensees must submit the following information to the NRC: certain facility changes for the NRC's approval; annual summaries of facility changes that did not need the NRC's preapproval; and annual updates to the ISA summaries.

Before the Subpart H requirements were implemented, the NRC relied on the 10-year license renewal as the main opportunity to review the facility safety basis. Now, along with the annual updates of the ISA summaries, the NRC is conducting more frequent reviews of the licensees' facility safety basis. Through the annual update of the ISA summaries, the NRC is kept informed of changes due to material degradation and aging throughout the lifetime of a facility. Thus, the Subpart H requirements permit the NRC to continue to support safe operations of licensed facilities on an ongoing basis, regardless of the duration of the license.

On August 24, 2006, the NRC staff provided the Commission with a paper, SECY–06–0186, 'Increasing Licensing Terms for Certain Fuel Cycle Facilities,' which recommended that the Commission approve a maximum license term of 40 years for certain fuel cycle facilities. The paper provided the basis for the staff's recommendation, including a description of the link with 10 CFR Part 70 reviews and a discussion of consistency with the NRC strategic goals for safety and effectiveness. In response to SECY–06–0186, the Commission issued a staff requirements memorandum (SRM) establishing the new policy described above. The Commission also approved of license terms for less than 40 years, on a case-by-case basis, where there are concerns with safety risk to the facility or in cases involving a new process or technology.

SECY–06–0186 and the SRM on SECY–06–0186 are available in the NRC's Public Document Room or electronically from the ADAMS Publicly Available Records (PARS) component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room).

FOR FURTHER INFORMATION CONTACT: Breeda Reilly, Project Manager, Fuel Manufacturing Branch, Fuel Facility Licensing Directorate, Division of Fuel Cycle Safety and Safeguards, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, DC 20555. Telephone: (301) 415–8103; fax: (301) 415–5955; e-mail: *bmr@nrc.gov.*

Dated at Rockville, Maryland this 21st day of November, 2006.

For the U.S. Nuclear Regulatory Commission.

Gary S. Janosko,

Deputy Director, Fuel Facility Licensing Directorate, Division of Fuel Cycle Safety and Safeguards, Office of Nuclear Material Safety and Safeguards.

[FR Doc. E6–20412 Filed 12–1–06; 8:45 am]

BILLING CODE 7590–01–P

OFFICE OF PERSONNEL MANAGEMENT

No FEAR Act Notice

AGENCY: Office of Personnel Management.

ACTION: Notice.

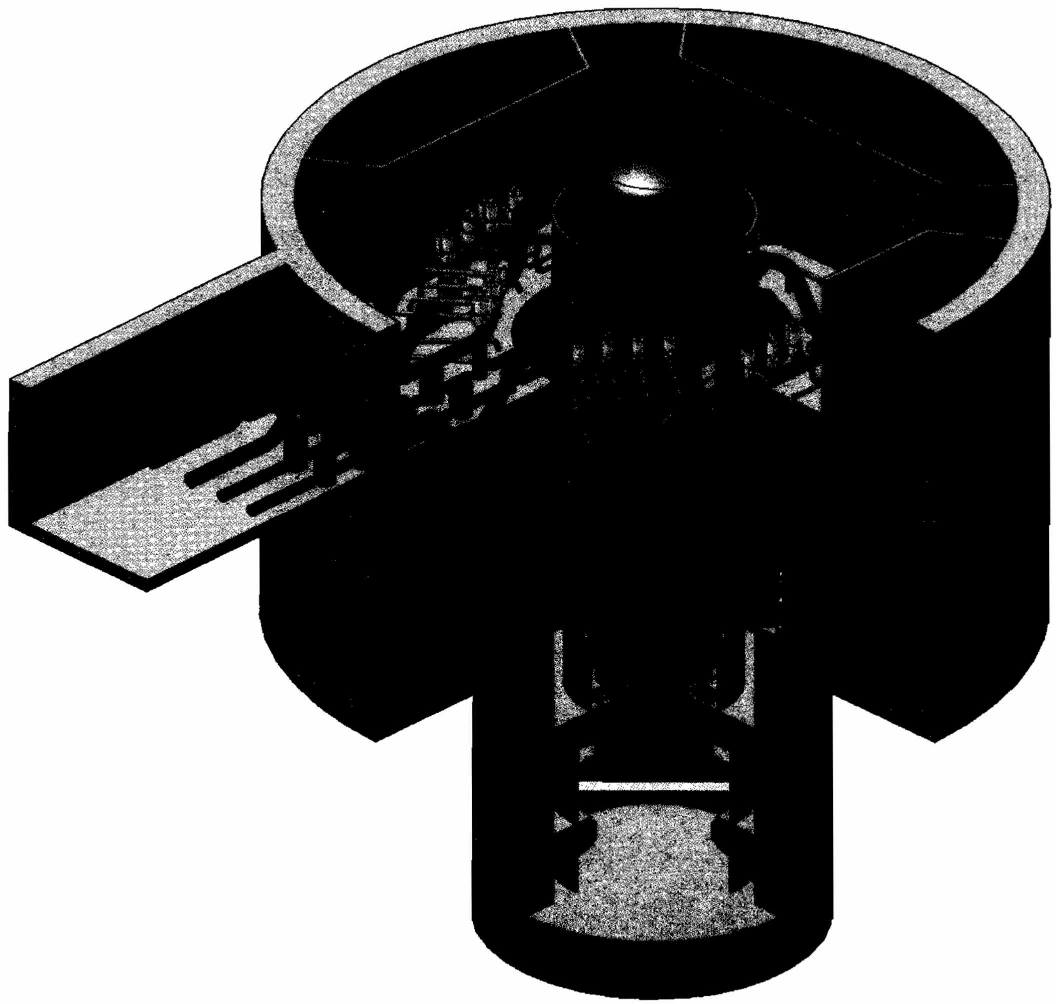
SUMMARY: Pursuant to 29 CFR part 724, the Office of Personnel Management (OPM) has implemented Title II of the Notification and Federal Employee Antidiscrimination and Retaliation Act (No FEAR Act) of 2002, concerning OPM's obligation (along with other Federal agencies) to provide notice to all its employees, former employees, and applicants for Federal employment about the rights and remedies available under the applicable Federal Antidiscrimination Laws and Whistleblower Protection Laws. OPM's No FEAR Act notice is available on OPM's Web site at *http://www.opm.gov/about_opm/nofear/.*

FOR FURTHER INFORMATION CONTACT: Stephen T. Shih, Chief, Center for Equal Employment Opportunity, by telephone at (202) 606–2460, by facsimile at (202) 606–1841, or by e-mail at *eeo@opm.gov.*

No FEAR Act Notice

On May 15, 2002, Congress enacted the "Notification and Federal Employee Antidiscrimination and Retaliation Act of 2002," which is now known as the No FEAR Act. One purpose of the Act

ESBWR Risk Management Overview



Presented By:
Rick Wachowiak
General Electric
December 14, 2006

imagination at work

GE Presentations in this Meeting

Overview of ESBWR & ESBWR PRA

Significant PRA Items for Revisions 1 and 2

Regulatory Treatment of Non-Safety Systems

Discussion of Previously Raised Issues

Background For This Meeting

Last Meeting Was in April 2006

Portions of PRA Had Been Revised

> Not All Members Had Seen Revised Parts

Further Dialog Determined To Be Necessary

Purpose of this Meeting

Outline Strategy for Risk Management in ESBWR Design

Discuss Dominant Accident Sequences Produced by the Analysis

Inform Subcommittee of Risk Reducing Design Changes That Have Been Implemented

Provide Update on RTNSS

Discuss Methods and Analyses Prior to Next Major Revision

ESBWR Basic Parameters

4,500 Megawatt Core Thermal Power

~1, 550 Megawatt Electric Gross

> Nominal Summer Rating

Natural Circulation

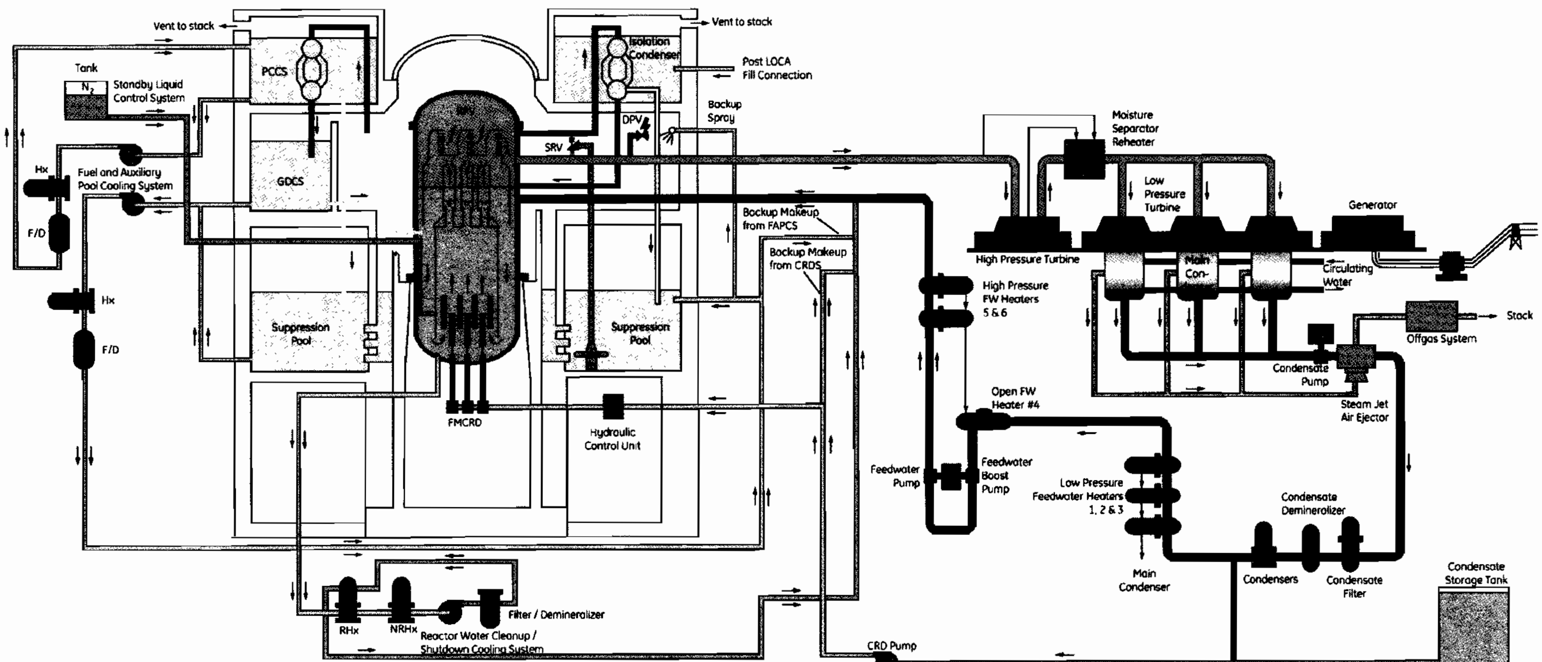
> No recirculation pumps

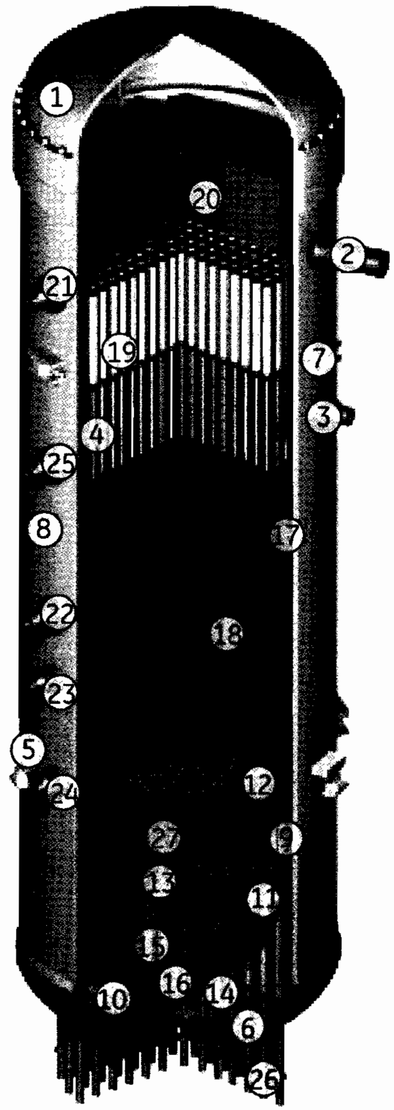
Passive Safety Systems

> No ECCS pumps

> 72 hours passive capability

ESBWR Overview

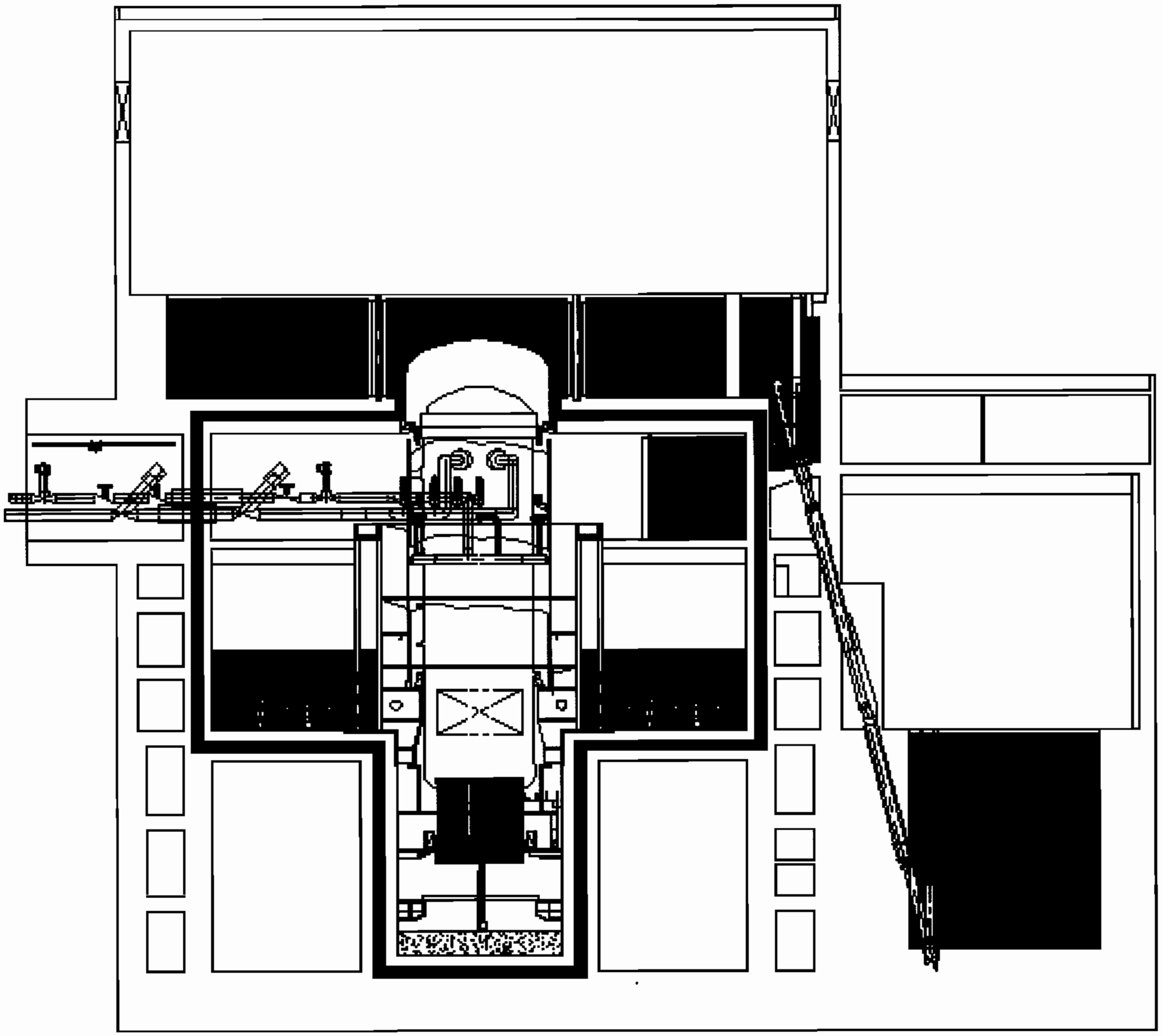




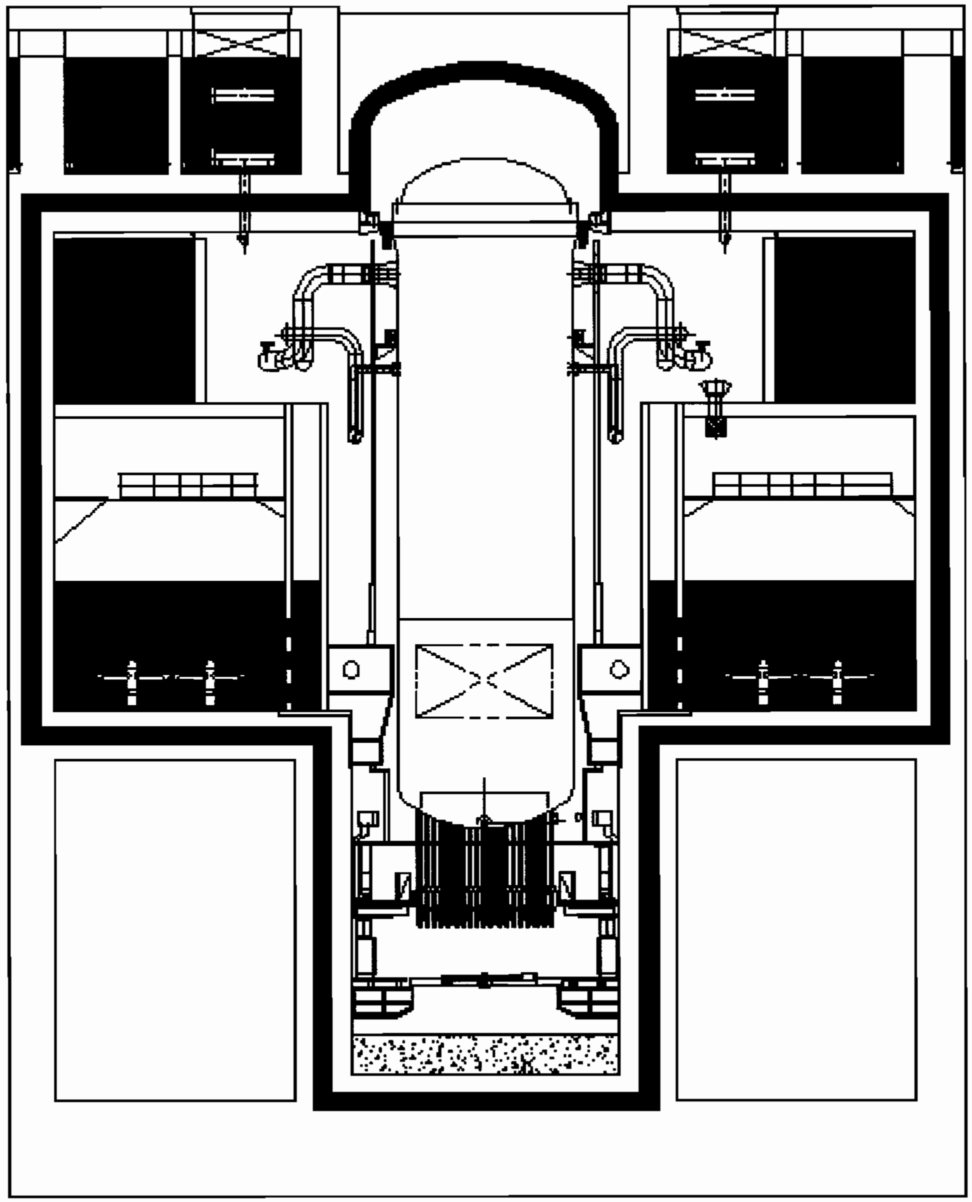
ESBWR

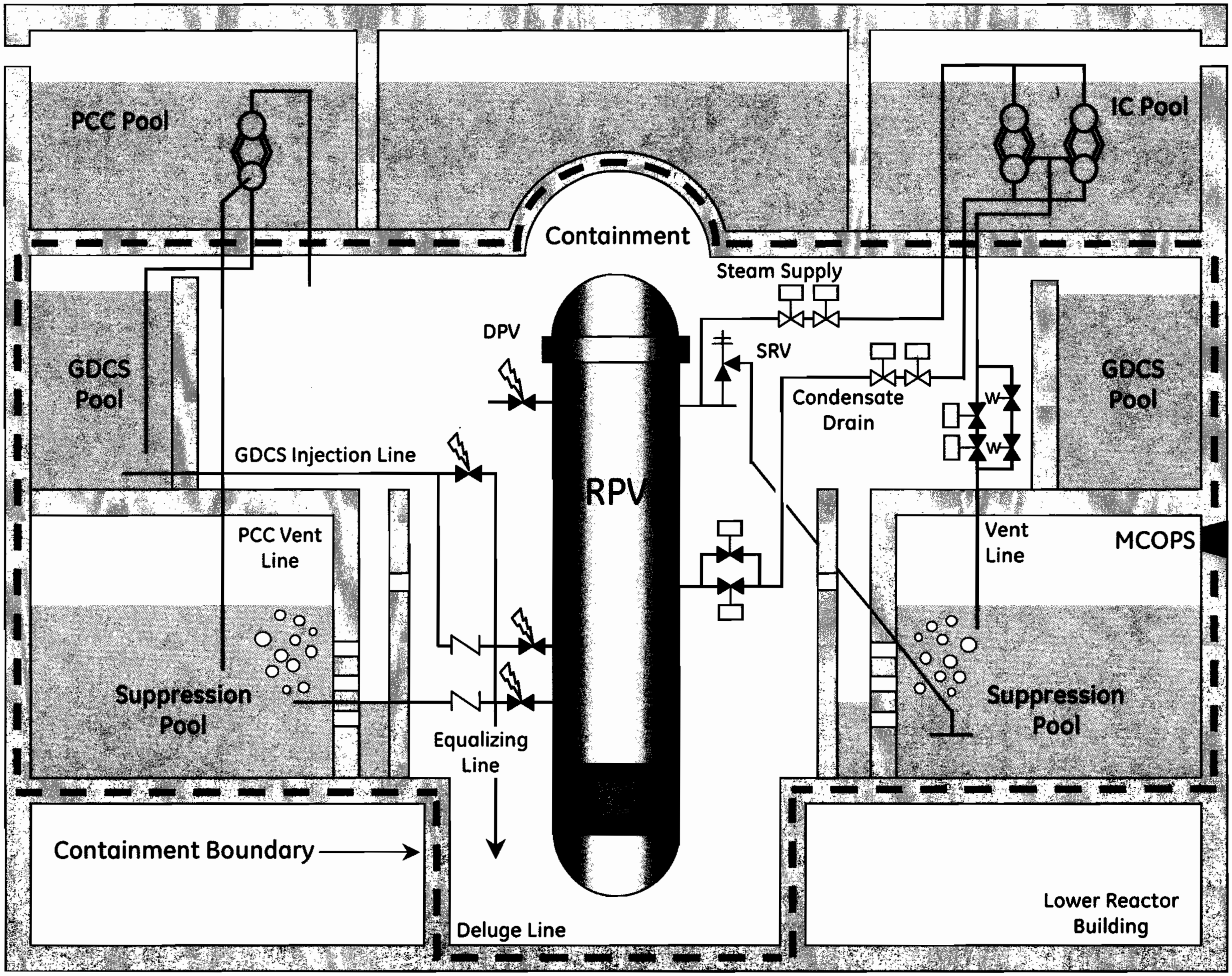
1. Vessel Flange and closure head
2. Steam outlet flow restrictor
3. Feedwater nozzle
4. Feedwater sparger
5. Vessel support
6. Vessel bottom head
7. Stabilizer
8. Forged shell rings
9. Core shroud
10. Shroud support brackets
11. Core plate
12. Top guide
13. Fuel supports
14. Control rod drive housings
15 Control rod guide tubes
16. In-core housing
17. Chimney
18. Chimney partitions
19. Steam separator assembly
20. Steam dryer assembly
21. DPV/IC outlet
22. IC return
23. GDCS inlet
24. GDCS equalizing line inlet
25. RWCU/SDC outlet
26. Control rod drives
27. Fuel and control rods

Reactor and Fuel Building

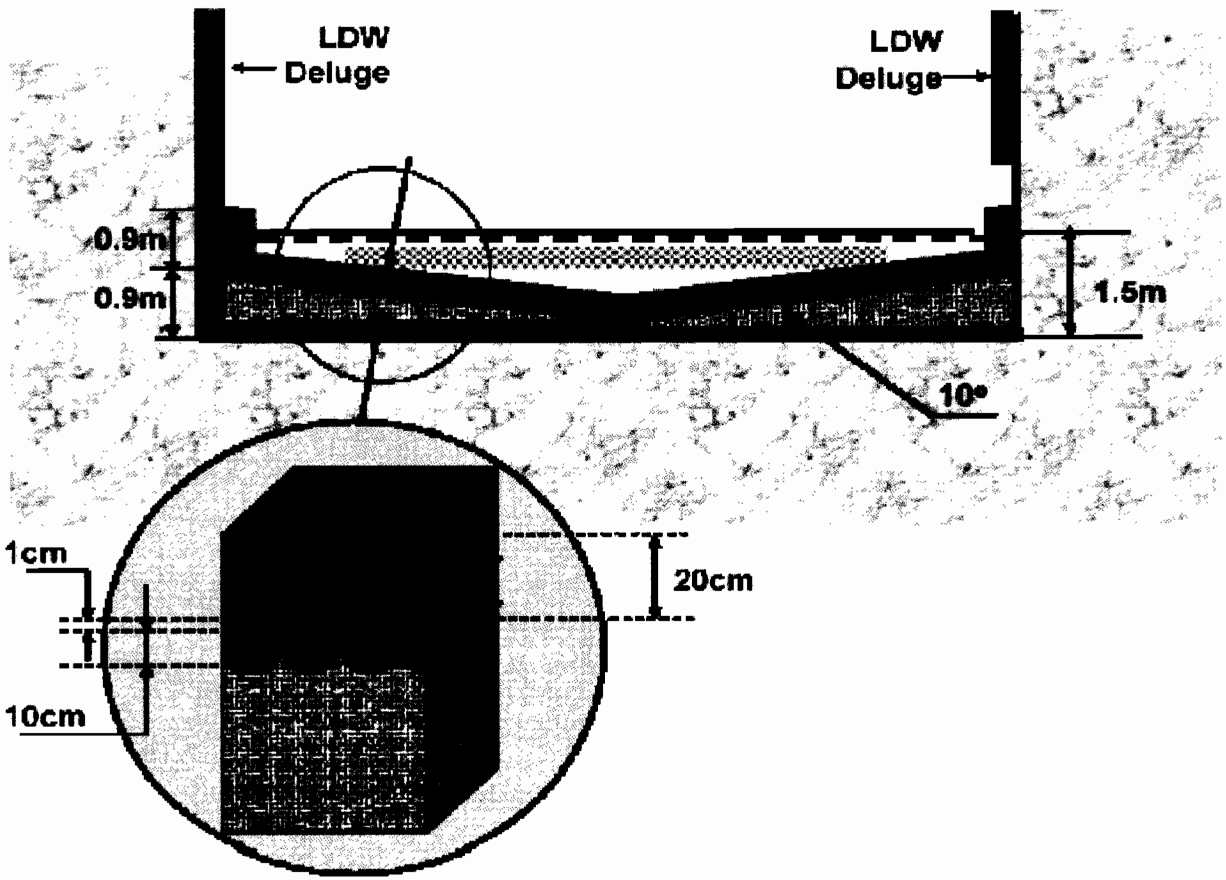


Containment

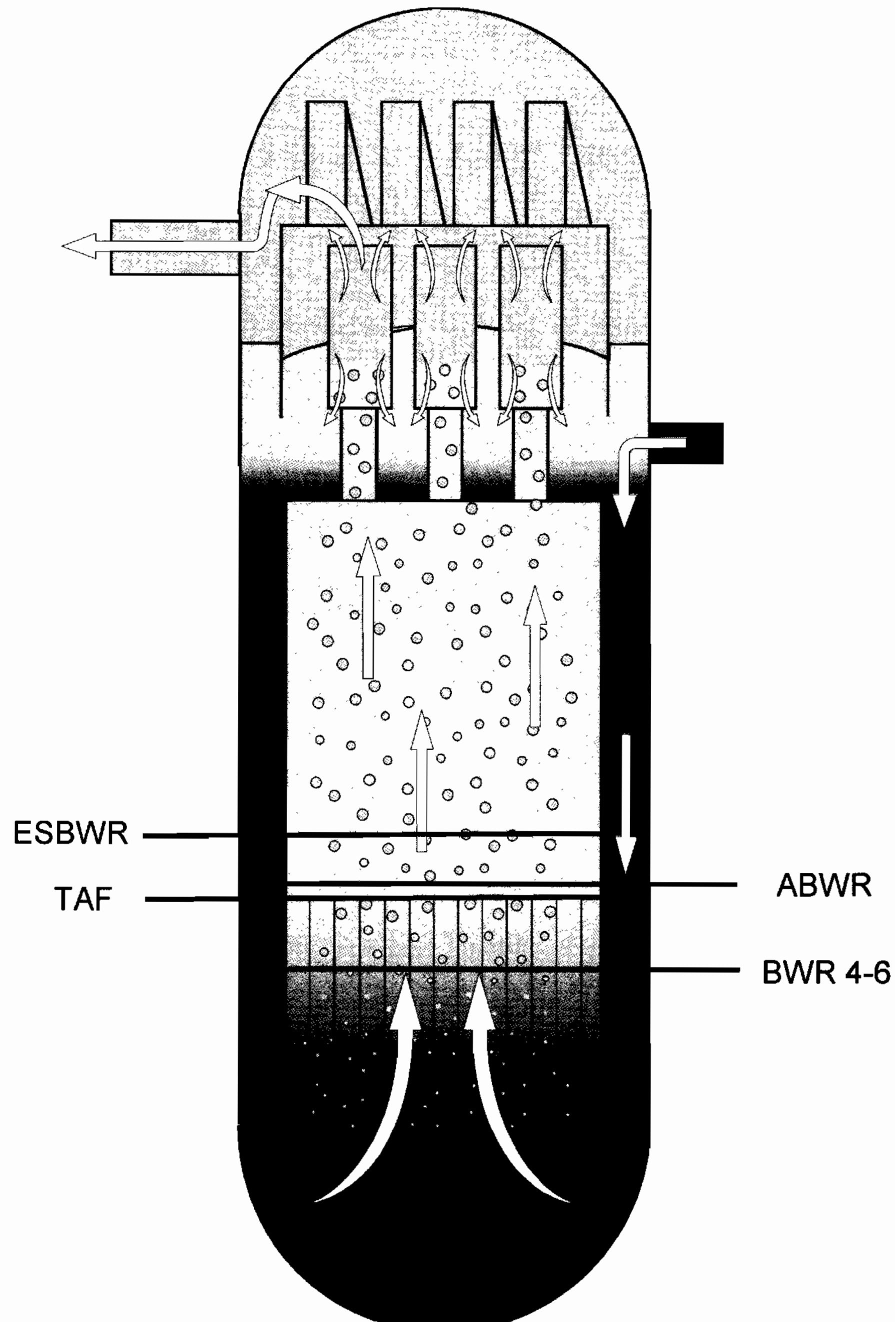



PCC Pool
IC Pool
Containment
Steam Supply
DPV
SRV
GDCS Pool
Condensate
Drain
GDCS
Pool
GDCS Injection Line
RPV
PCC Vent
Line
Vent
Line
MCOPS
Suppression
Pool
Suppression
Pool
Equalizing
Line
Containment Boundary
Deluge Line
Lower Reactor
Building

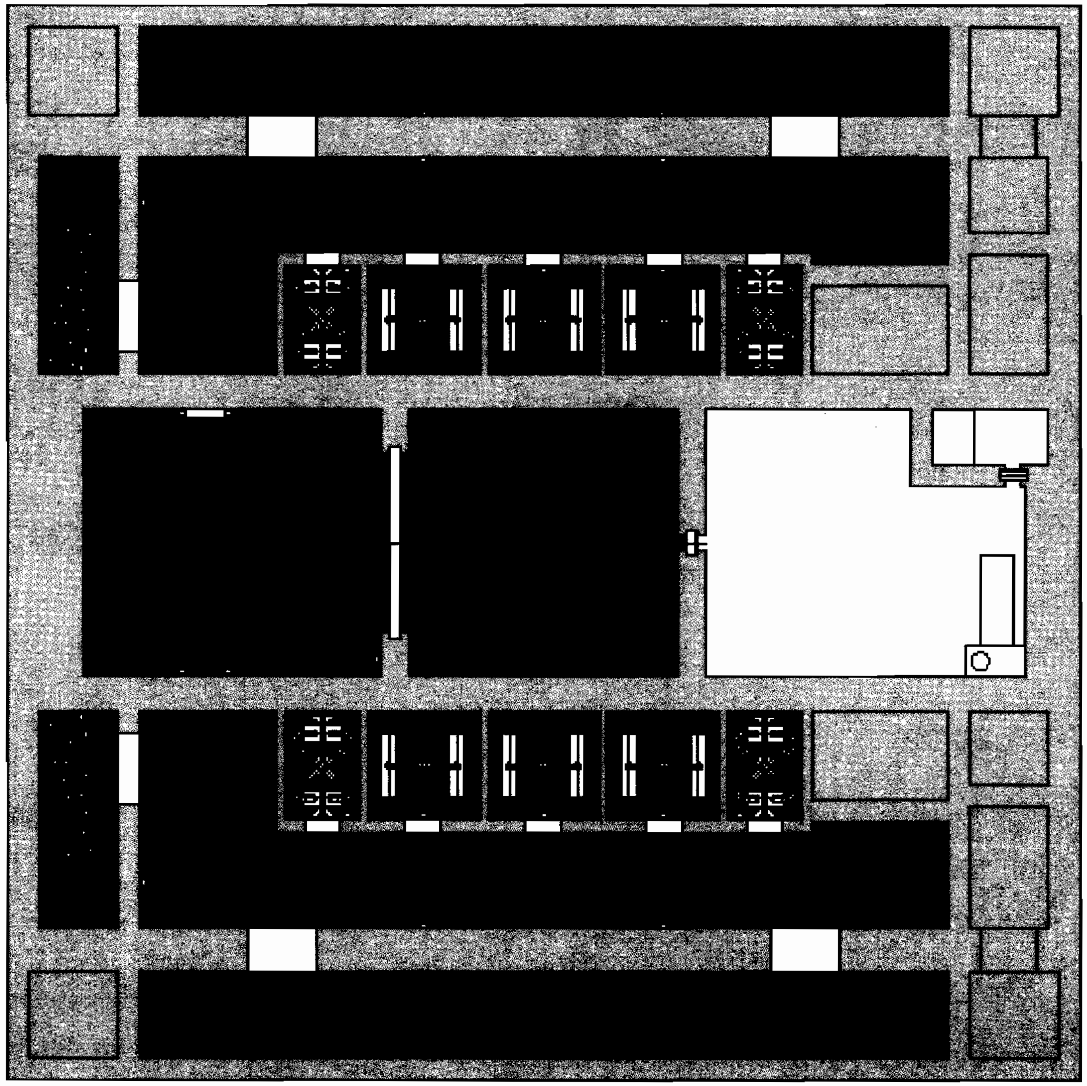
BiMAC Detail



LOCA Water Level Response



72 Hours Passive Capability



ESBWR Risk Management Program Goals
DCD Phase

Demonstrate that ESBWR Meets Established Risk Goals

Demonstrate that ESBWR Design Presents Lower Risk to Public than Existing Plants

Identify Non-Safety Functions Requiring Enhanced Regulatory Oversight

Systematic Search for Vulnerabilities

Extend Defense-in-Depth to Severe Accident Scenarios

Provide Framework for the Plant Specific PRA

Evolution of a Design and PRA

Conceptual Design	Design Base (DCD)	[redacted]	Construction Design	Plant in Operation
Is Design Feasible?	Can Design be Licensed?	[redacted]	Confirmation of Assumptions	Confirmation of Assumptions
Low Design Detail	Major Components Specified	[redacted]	All Components Described	All Components Described
Qualitative Risk Assessment	Qualitative & Quantitative PRA	[redacted]	Quantitative PRA with Fewer Gaps	As-Built As-Operated PRA
Defense-in-Depth Concepts	Defense-in-Depth Analyzed	[redacted]	No Defense-in-Depth Issues	No Defense-in-Depth Issues
Past Vulnerabilities Addressed	Sequence Level Vulnerabilities Eliminated	[redacted]	Component Level Vulnerabilities Eliminated	Additional Vulnerabilities Eliminated

← DCD / COLA Level of Design Detail

Scope of DCD PRA

Internal Events - Full Power

> Levels 1, 2, and 3

Internal Events – Shutdown

> Level 1 and Simplified Level 2

External Events

> Internal Fire (Bounding), Internal Flood, High Winds

> Seismic Margins

> Level 1

> Full Power and Shutdown

This Scope is Appropriate for ESBWR PRA Program Goals

Extended Defense – In - Depth

Classical Design / Analyses Provides DID using "Design Basis" Assumptions

ESBWR Adds Severe Accident Consideration

Main Objective is to Address Common Cause Failures
- > Historically Addressed by Additional Requirements on SSCs
- > ESBWR Adds Diversity to Design to Minimize Effect of CCF

Assessment of Non-Safety Equipment Performance Provided in Licensing Basis
- > Used to provide operational requirements (RTNSS)

PRA as a Design Tool

Overall Objective:

Eliminate Severe Accident Vulnerabilities

PRA Provides a Systematic Means for Finding and Eliminating These Vulnerabilities

Effectiveness May Be Limited By Information Availability Early in Design Phase

Easier to Make Corrections Earlier in Design Phase

Imperfect Tool is Better than None at All

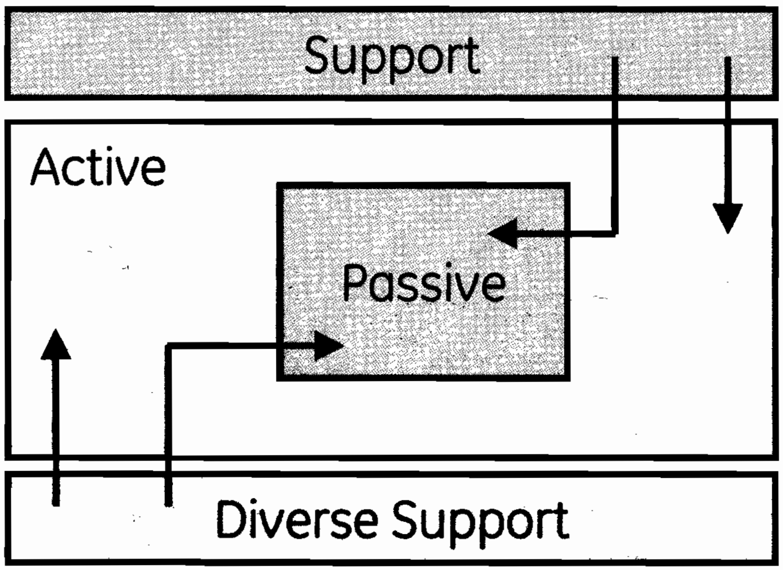
Key Features of ESBWR Risk Management

Passive Safety Systems

Active Asset Protection Systems

Support System Diversity

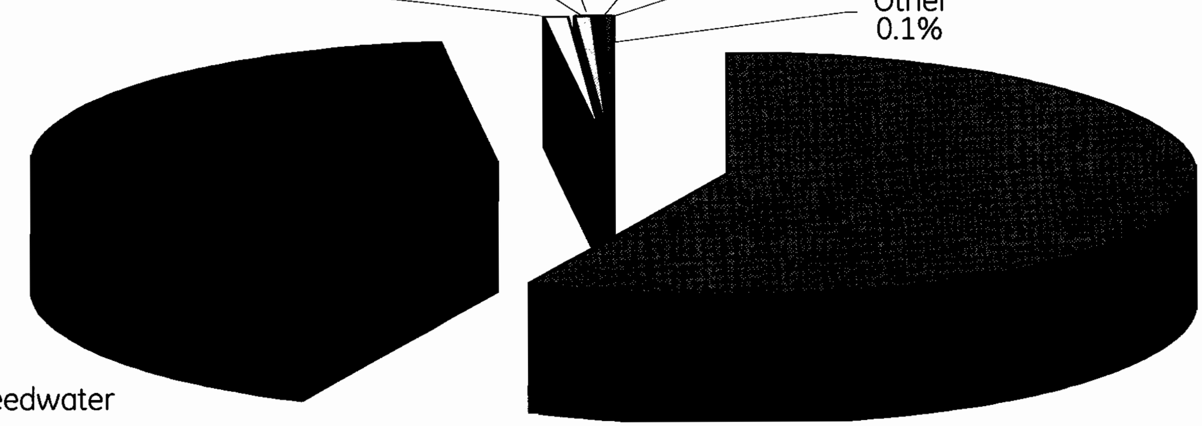
Target Configuration
for Core Damage
Prevention Functions



Functions for Core Damage Prevention

	Passive	Active
Reactivity Control	RPS SLCS	ARI FMCRD
Pressure Control	ICS SRV	Main Condenser
Inventory (High Press)	ICS	Feedwater CRD
Inventory (Low Press)	GDCS	FAPCS Fire Water Injection
Depressurization	DPV	SRV
Decay Heat Removal	PCCS ICS	Main Condenser RWCU

Level 1 Internal Events Results



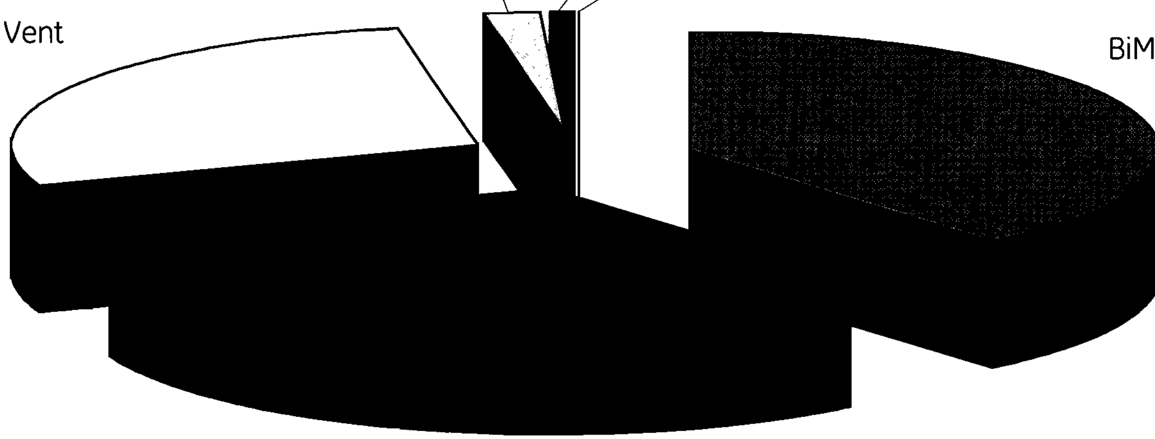
Total CDF

Point Estimate: 2.9×10^{-8}

Mean: 2.6×10^{-8}

95th: 8.3×10^{-8}

Level 2 Internal Events Breakdown



Total LRF

8×10^{-10}

This is Less Than 3% of CDF

External Events And Shutdown Results

Internal Events Shutdown

CDF: 5.6×10^{-9} LRF: Assumes no containment

Bounding Fire CDF

Power: 1.2×10^{-8} Shutdown: 2.3×10^{-8}

Flood CDF

Power: 3.7×10^{-9} Shutdown: 1.6×10^{-9}

High Wind CDF

Power and Shutdown: $< 10^{-10}$

When Detailed Design Is Considered, Fire and Flood Numbers Will Go Down

Conclusions

ESBWR Design is Robust

Probability of Severe Accident is Remote

Use of PRA as a Design Tool Ensured this Result

Combination of Passive Safety, Active Non-Safety Systems, and Diversity Leads to these Results

Table 7.2-5

Top Ten Level 1 Accident Sequences

Sequence	T-LOPP044 - Sequence No. 1	
CDF		1.63E-08
% of Class I CDF		57.07%
% of total CDF		55.91%
Initiating event	Loss of Preferred Power	

Scram is successful

Initial water drop causes level to go below L1.5

2 CRD Pumps fail to restore water before 15 minute timer expires, or the injection valves of more than 1 ICS train fail to open

ADS is successful

Depressurization causes ICS to be ineffective

Injection systems fail

Vessel fails at low pressure

Lower drywell water level is LOW

Table 7.2-5

Top Ten Level 1 Accident Sequences

Sequence	T-FDW044 – Sequence No. 2	
CDF		1.20E-08
% of Class I CDF		41.90%
% of total CDF		41.04%
Initiating event	Loss of Feedwater	

Scram is successful

Initial water drop causes level to go below L1.5

2 CRD Pumps fail to restore water before 15 minute timer expires, or the injection valves of more than 1 ICS train fail to open

ADS is successful

Depressurization causes ICS to be ineffective

Injection systems fail

Vessel fails at low pressure

Lower drywell water level is LOW

Table 7.2-5

Top Ten Level 1 Accident Sequences

Sequence	T-LOPP049 – Sequence No. 3	
CDF		3.19E-10
% of Class III CDF		77.64%
% of total CDF		1.09%
Initiating event	Loss of Preferred Power	

Scram is successful

Initial water drop causes level to go below L1.5

Failure to recover power within 15 minutes

2 CRD Pumps fail to restore water before 15 minute timer expires, or the injection valves of more than 1 ICS train fail to open

Depressurization fails

IC fails

SRVs prevent vessel overpressurization

CRD Pumps fails to provide injection

Operators fail to manually depressurize the plant

Vessel fails at high pressure

Table 7.2-5

Top Ten Level 1 Accident Sequences

Sequence	ML-L-014 – Sequence No. 4	
CDF		2.23E-10
% of Class I CDF		0.78%
% of total CDF		0.76%
Initiating event	Medium Liquid LOCA	

Scram is Successful
Vacuum breakers seat successfully
Feedwater fails to inject at high pressure
Fire Protection System Fails to Inject
ADS is successful
GDCS or equalizing lines fail to inject

FAPCS cannot provide long term injection because of insufficient water in suppression pool
CRD is not asked in this event tree. Inadequate water supply.
Lower drywell water level is HIGH

T-FDW	C	U2CISHORT	ADS	I	M	U1C	XSS	VLF	ADS	VG	VLFL
Loss of Feedwater Transient	(RPS or ARI) and CRM	2/2 CRD for 10 Minutes and 3/4 ICS	4 DPVs	3/4 IC	1/18 SRV	1/2 CRD	5 SRV	1/2 FAPCS or 1/1 FPS Injection	4 DPVs	2/8 lines and (2/3 GDCS pools or 1/3	1/2 FAPCS or 1/1 FPS Injection after ADS

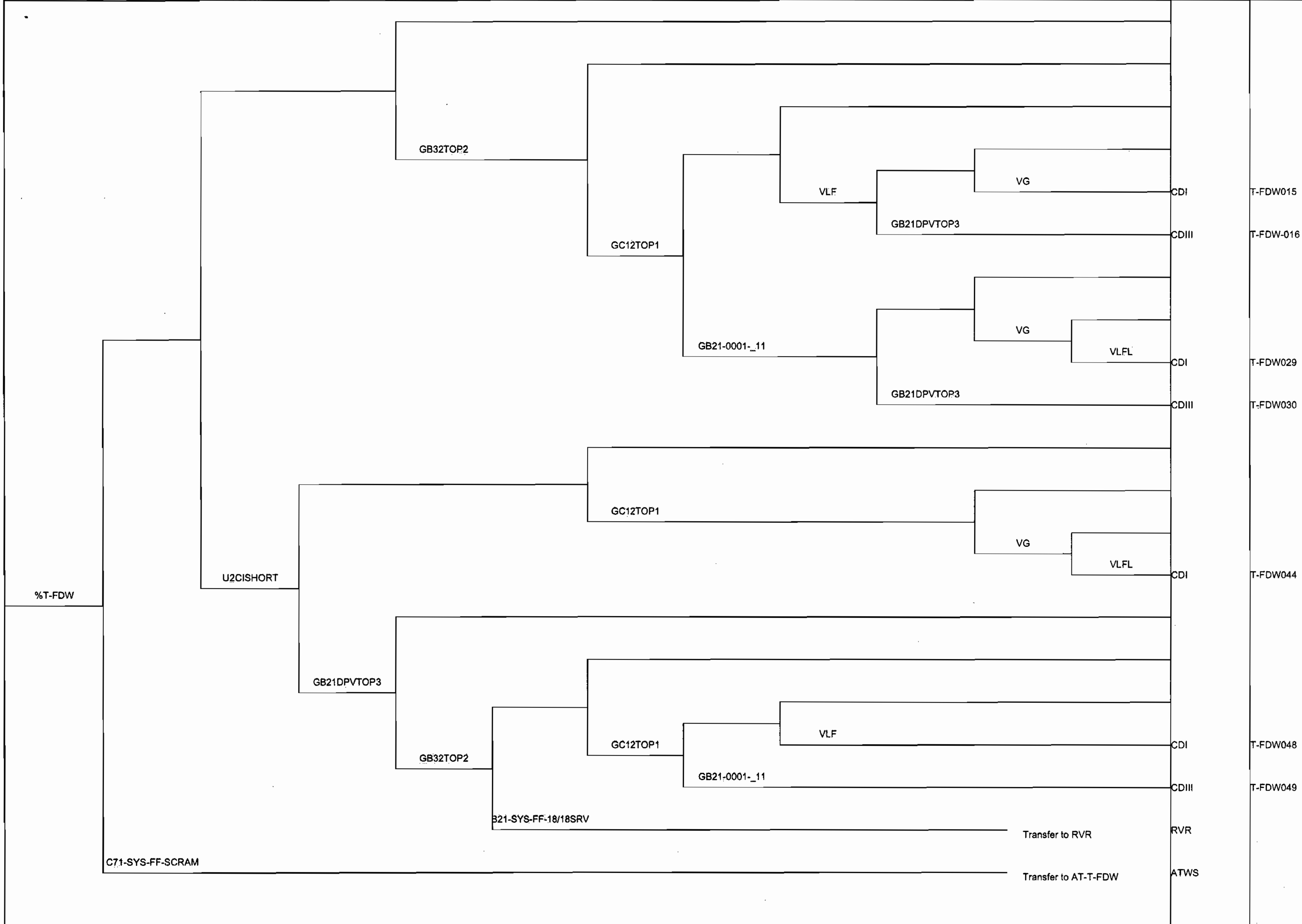


Table 7.2-7

Sequence 1 - Loss of Preferred Power Transient (T-LOPP044)

#	Cutset Probability	Event Probability	Event Name	Event Description
1	2.56E-10	4.60E-02	%T-LOPP	LOSS OF PREFERRED POWER TRANSIENT
		4.80E-02	C12-XHE-MH-F013A	MISPOSITION OF VALVE F013A
		4.80E-02	C12-XHE-MH-F013B	MISPOSITION OF VALVE F013B
		1.50E-05	E50-SQV-CF-GDCS7OPEN	CCF OF 7 SQUIB VALVES IN GDCS LINES TO OPEN
		1.61E-01	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP (LOCA)
2	2.56E-10	4.60E-02	%T-LOPP	LOSS OF PREFERRED POWER TRANSIENT
		4.80E-02	C12-XHE-MH-F013A	MISPOSITION OF VALVE F013A
		4.80E-02	C12-XHE-MH-F013B	MISPOSITION OF VALVE F013B
		1.50E-05	E50-SQV-CF-OPENALL	CCF OF ALL SQUIB VALVES TO OPEN
		1.61E-01	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP (LOCA)
3	2.56E-10	4.60E-02	%T-LOPP	LOSS OF PREFERRED POWER TRANSIENT
		4.80E-02	C12-XHE-MH-F013A	MISPOSITION OF VALVE F013A
		4.80E-02	C12-XHE-MH-F015B	MISPOSITION OF VALVE F015B
		1.50E-05	E50-SQV-CF-GDCS7OPEN	CCF OF 7 SQUIB VALVES IN GDCS LINES TO OPEN
		1.61E-01	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP (LOCA)
4	2.56E-10	4.60E-02	%T-LOPP	LOSS OF PREFERRED POWER TRANSIENT
		4.80E-02	C12-XHE-MH-F013A	MISPOSITION OF VALVE F013A
		4.80E-02	C12-XHE-MH-F015B	MISPOSITION OF VALVE F015B
		1.50E-05	E50-SQV-CF-OPENALL	CCF OF ALL SQUIB VALVES TO OPEN
		1.61E-01	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP (LOCA)
5	2.56E-10	4.60E-02	%T-LOPP	LOSS OF PREFERRED POWER TRANSIENT
		4.80E-02	C12-XHE-MH-F013B	MISPOSITION OF VALVE F013B
		4.80E-02	C12-XHE-MH-F015A	MISPOSITION OF VALVE F015A
		1.50E-05	E50-SQV-CF-GDCS7OPEN	CCF OF 7 SQUIB VALVES IN GDCS LINES TO OPEN
		1.61E-01	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP (LOCA)
6	2.56E-10	4.60E-02	%T-LOPP	LOSS OF PREFERRED POWER TRANSIENT
		4.80E-02	C12-XHE-MH-F013B	MISPOSITION OF VALVE F013B

Table 7.2-7

Sequence 1 - Loss of Preferred Power Transient (T-LOPP044)

#	Cutset Probability	Event Probability	Event Name	Event Description
		4.80E-02	C12-XHE-MH-F015A	MISPOSITION OF VALVE F015A
		1.50E-05	E50-SQV-CF-OPENALL	CCF OF ALL SQUIB VALVES TO OPEN
		1.61E-01	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP (LOCA)
7	2.56E-10	4.60E-02	%T-LOPP	LOSS OF PREFERRED POWER TRANSIENT
		4.80E-02	C12-XHE-MH-F015A	MISPOSITION OF VALVE F015A
		4.80E-02	C12-XHE-MH-F015B	MISPOSITION OF VALVE F015B
		1.50E-05	E50-SQV-CF-GDCS7OPEN	CCF OF 7 SQUIB VALVES IN GDCS LINES TO OPEN
		1.61E-01	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP (LOCA)
8	2.56E-10	4.60E-02	%T-LOPP	LOSS OF PREFERRED POWER TRANSIENT
		4.80E-02	C12-XHE-MH-F015A	MISPOSITION OF VALVE F015A
		4.80E-02	C12-XHE-MH-F015B	MISPOSITION OF VALVE F015B
		1.50E-05	E50-SQV-CF-OPENALL	CCF OF ALL SQUIB VALVES TO OPEN
		1.61E-01	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP (LOCA)
9	1.78E-10	4.60E-02	%T-LOPP	LOSS OF PREFERRED POWER TRANSIENT
		1.60E-03	B21-UV_-CC-F102B	CHECK VALVE #1 IN FEEDWATER LINE B FAILS TO REOPEN
		1.50E-05	E50-SQV-CF-GDCS7OPEN	CCF OF 7 SQUIB VALVES IN GDCS LINES TO OPEN
		1.61E-01	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP (LOCA)
10	1.78E-10	4.60E-02	%T-LOPP	LOSS OF PREFERRED POWER TRANSIENT
		1.60E-03	B21-UV_-CC-F102B	CHECK VALVE #1 IN FEEDWATER LINE B FAILS TO REOPEN
		1.50E-05	E50-SQV-CF-OPENALL	CCF OF ALL SQUIB VALVES TO OPEN
		1.61E-01	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP (LOCA)
11	1.78E-10	4.60E-02	%T-LOPP	LOSS OF PREFERRED POWER TRANSIENT
		1.60E-03	B21-UV_-CC-F103B	CHECK VALVE #2 IN FEEDWATER LINE B FAILS TO REOPEN
		1.50E-05	E50-SQV-CF-GDCS7OPEN	CCF OF 7 SQUIB VALVES IN GDCS LINES TO OPEN
		1.61E-01	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP (LOCA)
12	1.78E-10	4.60E-02	%T-LOPP	LOSS OF PREFERRED POWER TRANSIENT
		1.60E-03	B21-UV_-CC-F103B	CHECK VALVE #2 IN FEEDWATER LINE B FAILS TO REOPEN
		1.50E-05	E50-SQV-CF-OPENALL	CCF OF ALL SQUIB VALVES TO OPEN

Table 7.2-7

Sequence 1 - Loss of Preferred Power Transient (T-LOPP044)

#	Cutset Probability	Event Probability	Event Name	Event Description
		1.61E-01	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP (LOCA)
13	1.78E-10	4.60E-02	%T-LOPP	LOSS OF PREFERRED POWER TRANSIENT
		1.60E-03	C12-UV_-CC-F022	CHECK VALVE F022 FAILS TO OPEN
		1.50E-05	E50-SQV-CF-GDCS7OPEN	CCF OF 7 SQUIB VALVES IN GDCS LINES TO OPEN
		1.61E-01	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP (LOCA)
14	1.78E-10	4.60E-02	%T-LOPP	LOSS OF PREFERRED POWER TRANSIENT
		1.60E-03	C12-UV_-CC-F022	CHECK VALVE F022 FAILS TO OPEN
		1.50E-05	E50-SQV-CF-OPENALL	CCF OF ALL SQUIB VALVES TO OPEN
		1.61E-01	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP (LOCA)
15	7.63E-11	4.60E-02	%T-LOPP	LOSS OF PREFERRED POWER TRANSIENT
		4.80E-02	C12-XHE-MH-F013A	MISPOSITION OF VALVE F013A
		4.80E-02	C12-XHE-MH-F013B	MISPOSITION OF VALVE F013B
		1.50E-05	E50-SQV-CF-GDCS7OPEN	CCF OF 7 SQUIB VALVES IN GDCS LINES TO OPEN
		4.80E-02	G21-XHE-MH-F334	MISPOSITION OF VALVE F334
16	7.63E-11	4.60E-02	%T-LOPP	LOSS OF PREFERRED POWER TRANSIENT
		4.80E-02	C12-XHE-MH-F013A	MISPOSITION OF VALVE F013A
		4.80E-02	C12-XHE-MH-F013B	MISPOSITION OF VALVE F013B
		1.50E-05	E50-SQV-CF-OPENALL	CCF OF ALL SQUIB VALVES TO OPEN
		4.80E-02	G21-XHE-MH-F334	MISPOSITION OF VALVE F334
17	7.63E-11	4.60E-02	%T-LOPP	LOSS OF PREFERRED POWER TRANSIENT
		4.80E-02	C12-XHE-MH-F013A	MISPOSITION OF VALVE F013A
		4.80E-02	C12-XHE-MH-F015B	MISPOSITION OF VALVE F015B
		1.50E-05	E50-SQV-CF-GDCS7OPEN	CCF OF 7 SQUIB VALVES IN GDCS LINES TO OPEN
		4.80E-02	G21-XHE-MH-F334	MISPOSITION OF VALVE F334
18	7.63E-11	4.60E-02	%T-LOPP	LOSS OF PREFERRED POWER TRANSIENT
		4.80E-02	C12-XHE-MH-F013A	MISPOSITION OF VALVE F013A
		4.80E-02	C12-XHE-MH-F015B	MISPOSITION OF VALVE F015B
		1.50E-05	E50-SQV-CF-OPENALL	CCF OF ALL SQUIB VALVES TO OPEN

Table 7.2-7

Sequence 1 - Loss of Preferred Power Transient (T-LOPP044)

#	Cutset Probability	Event Probability	Event Name	Event Description
		4.80E-02	G21-XHE-MH-F334	MISPOSITION OF VALVE F334
19	7.63E-11	4.60E-02	%T-LOPP	LOSS OF PREFERRED POWER TRANSIENT
		4.80E-02	C12-XHE-MH-F013B	MISPOSITION OF VALVE F013B
		4.80E-02	C12-XHE-MH-F015A	MISPOSITION OF VALVE F015A
		1.50E-05	E50-SQV-CF-GDCS7OPEN	CCF OF 7 SQUIB VALVES IN GDCS LINES TO OPEN
		4.80E-02	G21-XHE-MH-F334	MISPOSITION OF VALVE F334
20	7.63E-11	4.60E-02	%T-LOPP	LOSS OF PREFERRED POWER TRANSIENT
		4.80E-02	C12-XHE-MH-F013B	MISPOSITION OF VALVE F013B
		4.80E-02	C12-XHE-MH-F015A	MISPOSITION OF VALVE F015A
		1.50E-05	E50-SQV-CF-OPENALL	CCF OF ALL SQUIB VALVES TO OPEN
		4.80E-02	G21-XHE-MH-F334	MISPOSITION OF VALVE F334
21	7.63E-11	4.60E-02	%T-LOPP	LOSS OF PREFERRED POWER TRANSIENT
		4.80E-02	C12-XHE-MH-F015A	MISPOSITION OF VALVE F015A
		4.80E-02	C12-XHE-MH-F015B	MISPOSITION OF VALVE F015B
		1.50E-05	E50-SQV-CF-GDCS7OPEN	CCF OF 7 SQUIB VALVES IN GDCS LINES TO OPEN
		4.80E-02	G21-XHE-MH-F334	MISPOSITION OF VALVE F334
22	7.63E-11	4.60E-02	%T-LOPP	LOSS OF PREFERRED POWER TRANSIENT
		4.80E-02	C12-XHE-MH-F015A	MISPOSITION OF VALVE F015A
		4.80E-02	C12-XHE-MH-F015B	MISPOSITION OF VALVE F015B
		1.50E-05	E50-SQV-CF-OPENALL	CCF OF ALL SQUIB VALVES TO OPEN
		4.80E-02	G21-XHE-MH-F334	MISPOSITION OF VALVE F334
23	6.81E-11	4.60E-02	%T-LOPP	LOSS OF PREFERRED POWER TRANSIENT
		1.50E-05	E50-SQV-CF-GDCS7OPEN	CCF OF 7 SQUIB VALVES IN GDCS LINES TO OPEN
		6.13E-01	R11-SYS-FF-NOREC	FAILURE IN OFFSITE POWER RECOVERY
		1.00E-03	R16-BT _-TM-R16BTA2	BATTERY R16-BTA2 IN TEST
		1.61E-01	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP (LOCA)
24	6.81E-11	4.60E-02	%T-LOPP	LOSS OF PREFERRED POWER TRANSIENT
		1.50E-05	E50-SQV-CF-GDCS7OPEN	CCF OF 7 SQUIB VALVES IN GDCS LINES TO OPEN

Table 7.2-7

Sequence 1 - Loss of Preferred Power Transient (T-LOPP044)

#	Cutset Probability	Event Probability	Event Name	Event Description
		6.13E-01	R11-SYS-FF-NOREC	FAILURE IN OFFSITE POWER RECOVERY
		1.00E-03	R16-BT_-TM-R16BTB2	BATTERY R16-BTB2 IN TEST
		1.61E-01	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP (LOCA)
25	6.81E-11	4.60E-02	%T-LOPP	LOSS OF PREFERRED POWER TRANSIENT
		1.50E-05	E50-SQV-CF-OPENALL	CCF OF ALL SQUIB VALVES TO OPEN
		6.13E-01	R11-SYS-FF-NOREC	FAILURE IN OFFSITE POWER RECOVERY
		1.00E-03	R16-BT_-TM-R16BTA2	BATTERY R16-BTA2 IN TEST
		1.61E-01	XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP (LOCA)

Table 11-1

Results of the 72-Hour Mission Time Sensitivity Analysis

Initiating Event	Quantification Results without CCV Treatment				Quantification Results after CCV Treatment			
	Acc. Sequence	CDF [/yr]	% of Class	% of Total	Acc. Sequence	CDF [/yr]	% of Class	% of Total
T-GEN	T-GEN022	4.22E-13	0.02%	0.00%	T-CCV02	3.13E-11	73.46%	0.11%
	T-GEN025	1.97E-13	0.01%	0.00%				
T-FDW	T-FDW039	3.27E-11	1.30%	0.10%				
	T-FDW042	8.72E-13	0.03%	0.00%				
T-IORV	T-IORV023	4.86E-12	0.19%	0.02%				
	T-IORV010	3.01E-13	0.01%	0.00%	T-IORV010	3.01E-13	0.71%	0.00%
T-SW	T-SW005	4.58E-12	0.18%	0.01%	T-SW005	4.58E-12	10.75%	0.02%
	T-SW007	1.20E-12	0.05%	0.00%	T-SW007	1.20E-12	2.81%	0.00%
	T-SW014	6.38E-12	0.25%	0.02%	T-SWCCV02	4.04E-13	0.95%	0.00%
	T-SW016	2.68E-12	0.11%	0.01%				
T-LOPP	T-LOPP008	2.20E-13	0.01%	0.00%	T-LOPPACCV02	ε		
	T-LOPP011	1.03E-13	0.00%	0.00%				
	T-LOPP021	3.92E-12	0.16%	0.01%	T-LOPPBCCV03	4.82E-12	11.33%	0.02%
	T-LOPP024	1.83E-12	0.07%	0.01%				
	T-LOPP036	1.51E-11	0.60%	0.05%				
	T-LOPP039	1.13E-09	45.00%	3.56%				
	T-LOPP042	1.94E-10	7.74%	0.61%				
LL-S	LL-S-015	9.99E-10	39.79%	3.15%	LL-S-CCV05	ε		
LL-S-FDWA	LL-S-FDWA014	1.11E-11	0.44%	0.04%	LL-S-FDWACCV04	ε		
LL-S-FDWB	LL-S-FDWB014	1.11E-11	0.44%	0.04%	LL-S-FDWBCCV04	ε		
ML-L	ML-L-016	7.53E-11	3.00%	0.24%	ML-L-CCV05	ε		
ML-L-RWCU	ML-L-RWCU015	1.47E-11	0.58%	0.05%	ML-L-RWCUCCV04	ε		
Class II Totals:		**2.51E-09**	100.00%	**7.91%**		**4.26E-11**	100.00%	**0.15%**
Total CDF, without Class II:		**2.92E-08**				**2.92E-08**		
Total CDF, including Class II:		**3.17E-08**				**2.93E-08**		

Note: "ε" indicates that the accident sequence quantification did not produce any cutsets above the truncation limit of 1.0E-13

T-GEN	C	TPCS	I	M	U1CF	XS5	VLF	ADS	
General Transient	(RPS or ARI) and CRM	BYPASS and 1/4 FDW	3/4 IC	1/18 SRV	1/2 CRD or 1/2 FDW	5 SRV	1/2 FAPCS or 1/1 FPS injection	4 DPVs	2/8 I GD 1/3 and

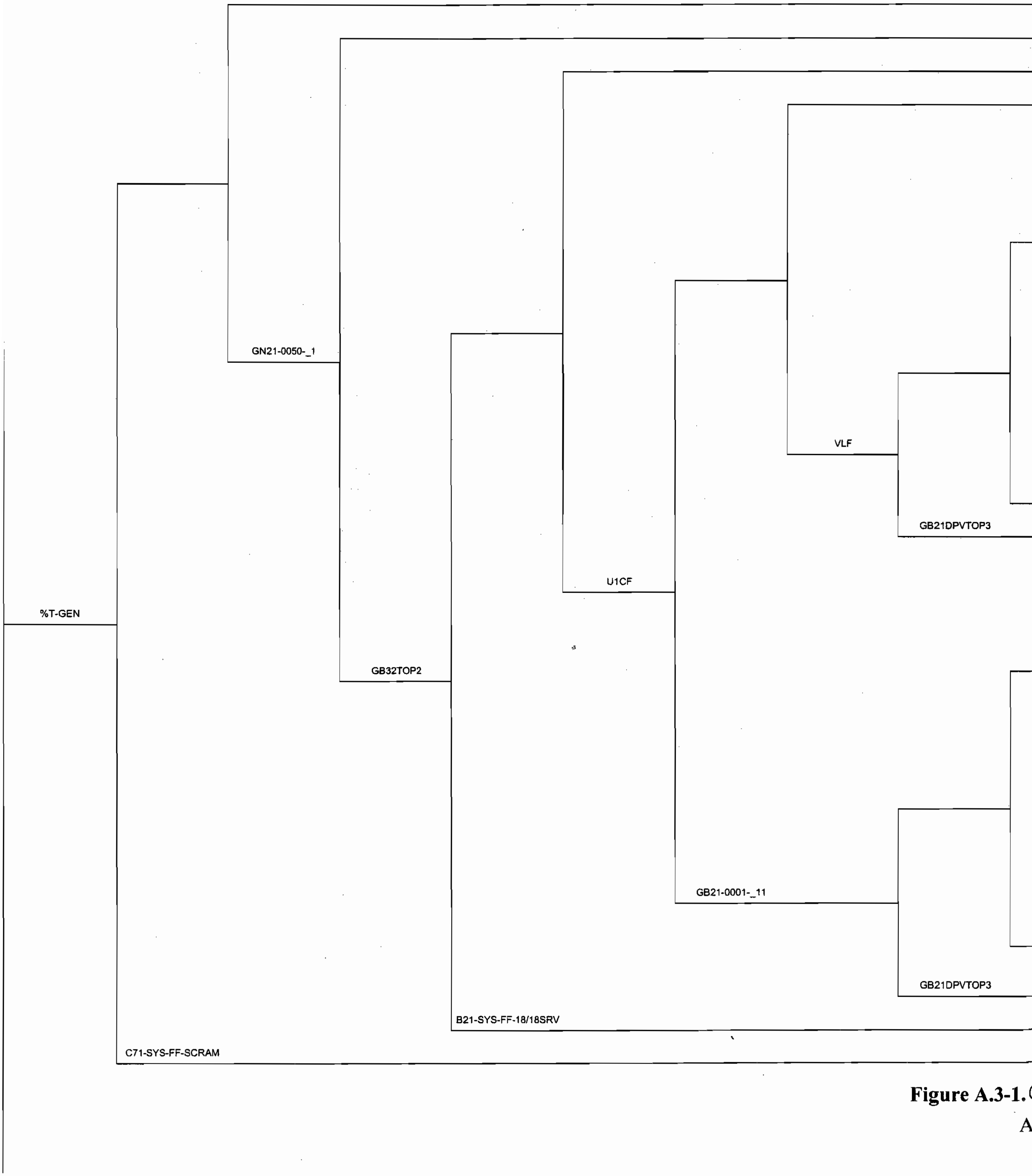
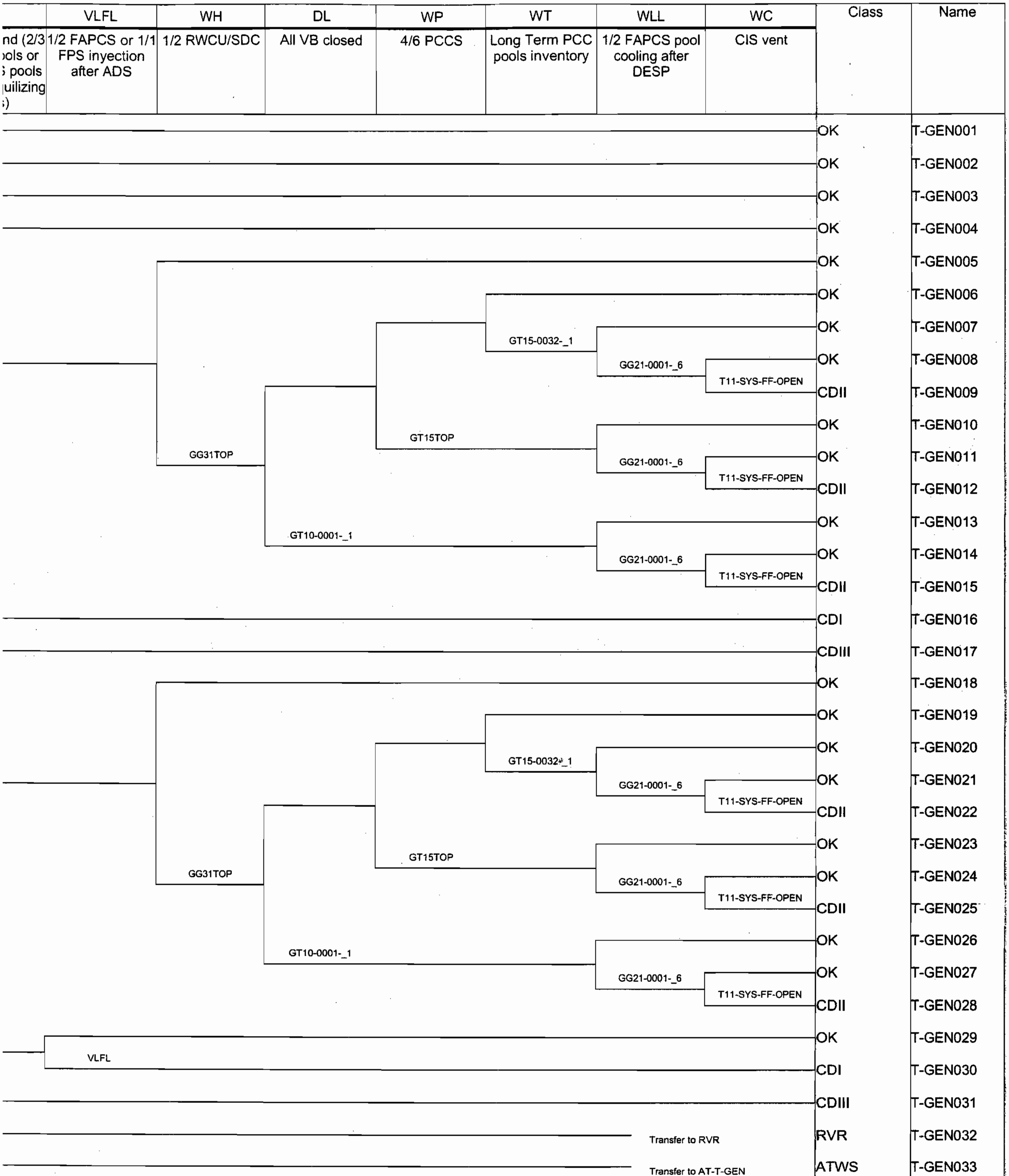


Figure A.3-1. G
A3



eneral Transient

-2

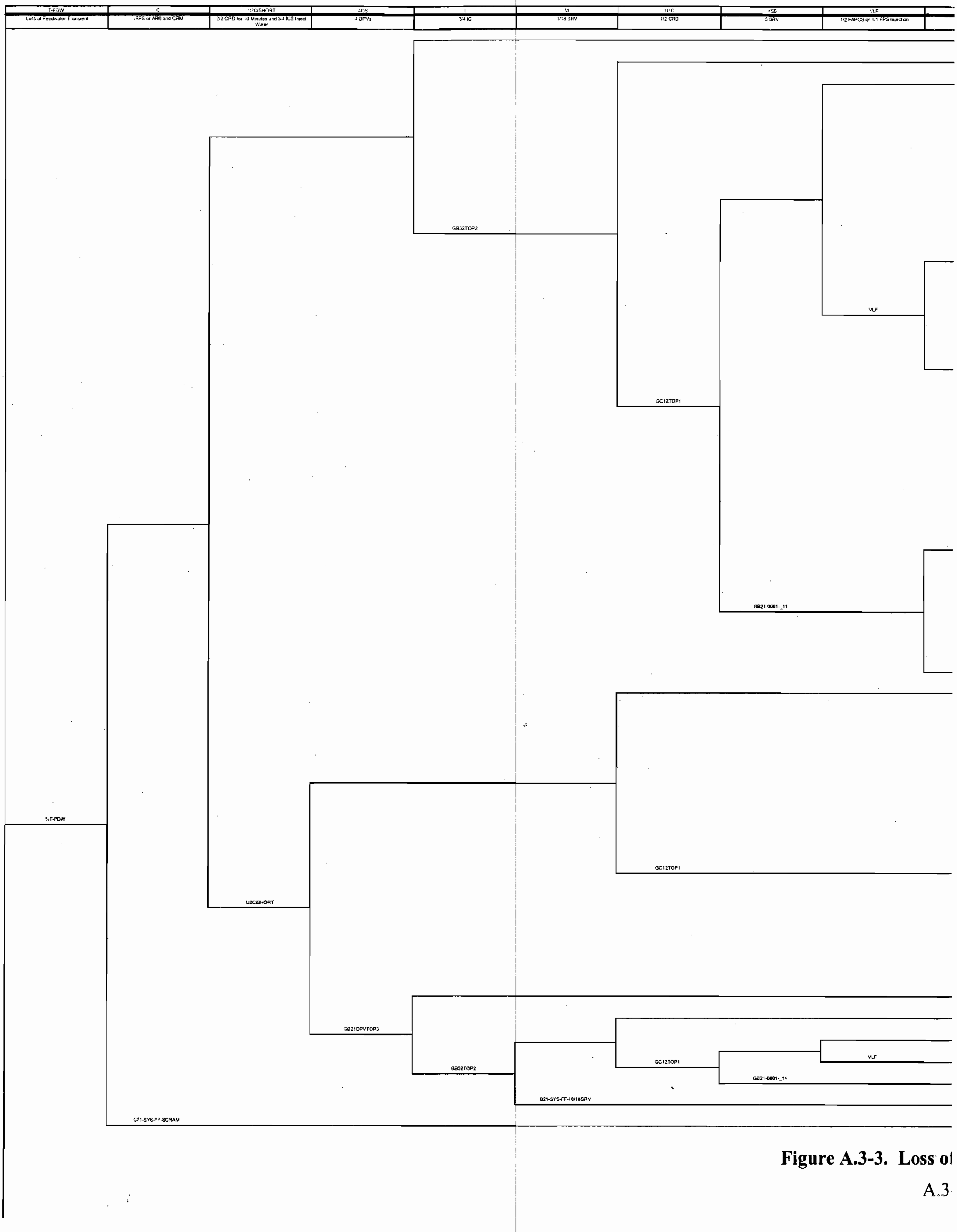
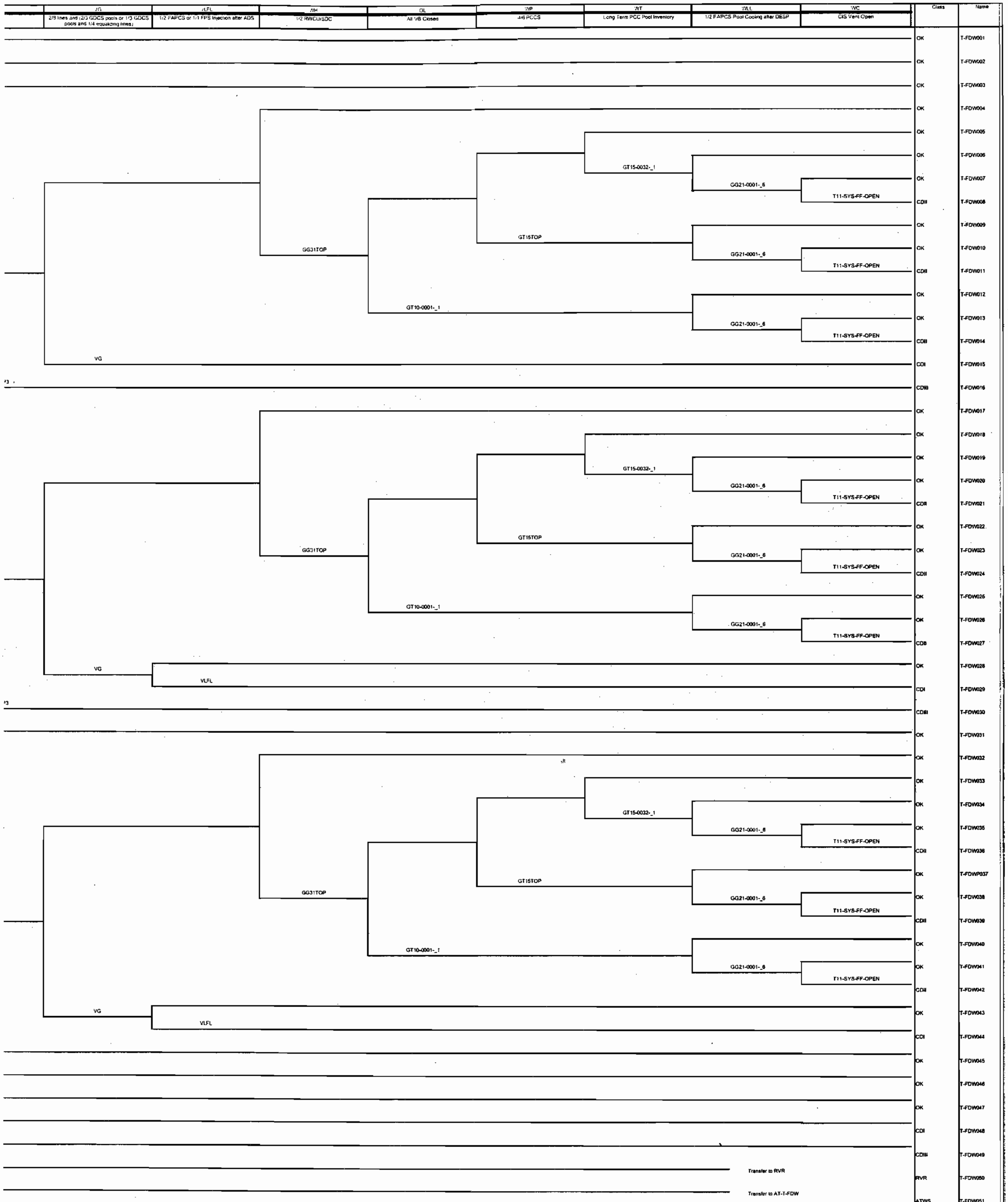


Figure A.3-3. Loss of

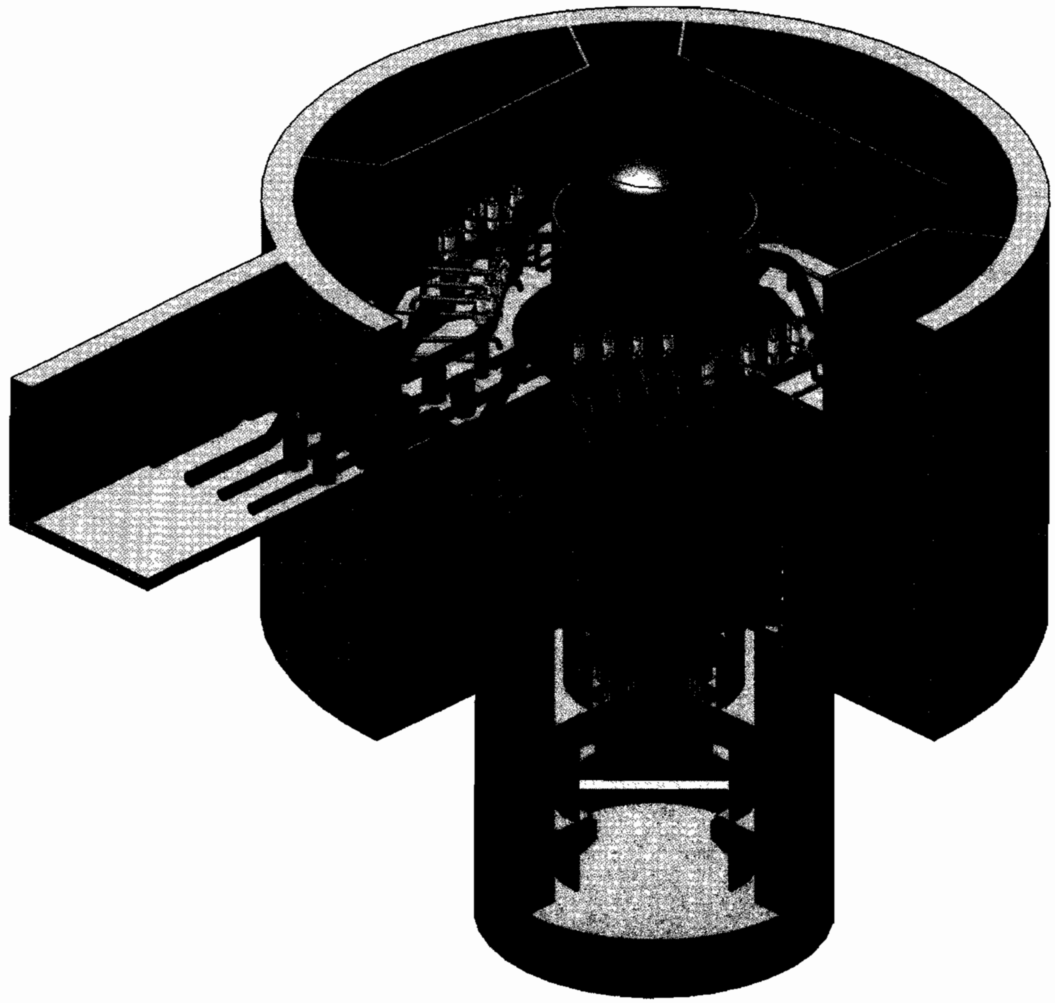


Feedwater Transient

NEDO-33201

-5

Probabilistic Risk Assessment



Update and Information Exchange

Presented By:
Rick Wachowiak
General Electric
December 14, 2006

imagination at work

GE Presentation Topics

PRA Rev 2 Scope

Affect of Major Design Changes

Scope of PRA Revision 2 – Base Model

Isolation Condenser Additional Water Volume

I&C Architecture and Requirements

Additional Design Detail for BOP Systems

Revise Common Cause Model (MGL Method)

Move Model Detail to Event Trees

Eliminate Sequence Specific Logic Flags

Add Model Detail for 24 – 72 Hour Sequences

Reconcile Component Names With DCD

Include Shutdown and Class II Sequences in LRF

Other Design Details as Information Becomes Available

Scope of PRA Revision 2 - Other

Basic Event Naming Convention May Change

> Required to implement URD database

> Eliminates patch in rev 1 model

Gate Names in Rev 1 Did Not Match Our Convention

Considering Direct Connection of CET to Level 1

ICS Design Change

Add Volume to ICS Condensate Return Lines

> 9 m^3 per IC

Allows Optimization of the Level 1 ECCS Signals

CRD Not Needed to Prevent Depressurization in Loss of Feedwater Events

Top 90% of Cutsets in Rev 1 Involve Loss of Feedwater + Loss of CRD

ICS Design Change – Additional Figures

ICS P&ID	Figure 5.1-3 Rev 1
	Figure 5.1-3 Rev 2
LOFW Water Level	Figure 15.2-16c Rev 1
	Figure 15.2-16c Rev 2
TGEN Event Tree	Figure A.3-1 Rev 1
TFDW Event Tree	Figure A.3-3 Rev 1

Digital Control and Instrumentation System Architecture

GE Has Chosen DCIS Architecture

Determined Diversity & Defense-In-Depth Requirements

Implementation of Safety-Related DCIS is Double Failure Proof

> Allows maintenance of 1 division without AOT

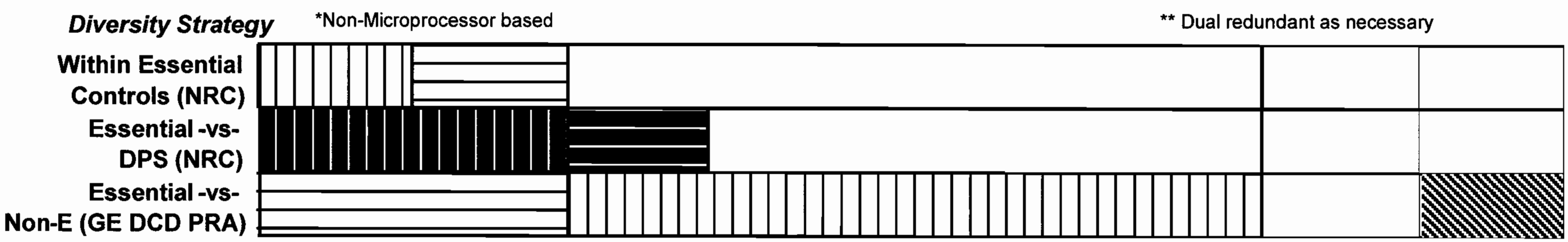
> OOS Time Controlled By TRM + Maintenance Rule

Diversity & Defense-In-Depth Strategy

Safety Category	Safety-Related E - DCIS		Nonsafety-Related NE - DCIS				
System Families	RPS NMS	ECCS ESF	DPS	NUCLEAR CONTROL SYSTEMS	Balance of any NE-DCIS Systems	PCS	Severe Accident
Architecture	NUMAC Derived	Redundant	Triple Redundant	Triple Redundant	Dual Redundant	Workstations **	PLCs
Systems	RPS LD&IS (MSIV) NMS ATWS/SLCS*	ICS SRV/DPV GDCS LD&IS (Non-MSIV)	RPS ECCS Backup	FWC, PAS (Automation) SB&PC, T/G Control	PIP A, PIP B Balance Of Plant (Power Generation)	HMI, Alarms, SPDF, Historian, 3D-Monicore	Deluge System

*Non-Microprocessor based

** Dual redundant as necessary



Check NEDO-33251 for Latest Version of this Figure

Robust DCIS Implementation

Double Failure Proof (N-2)

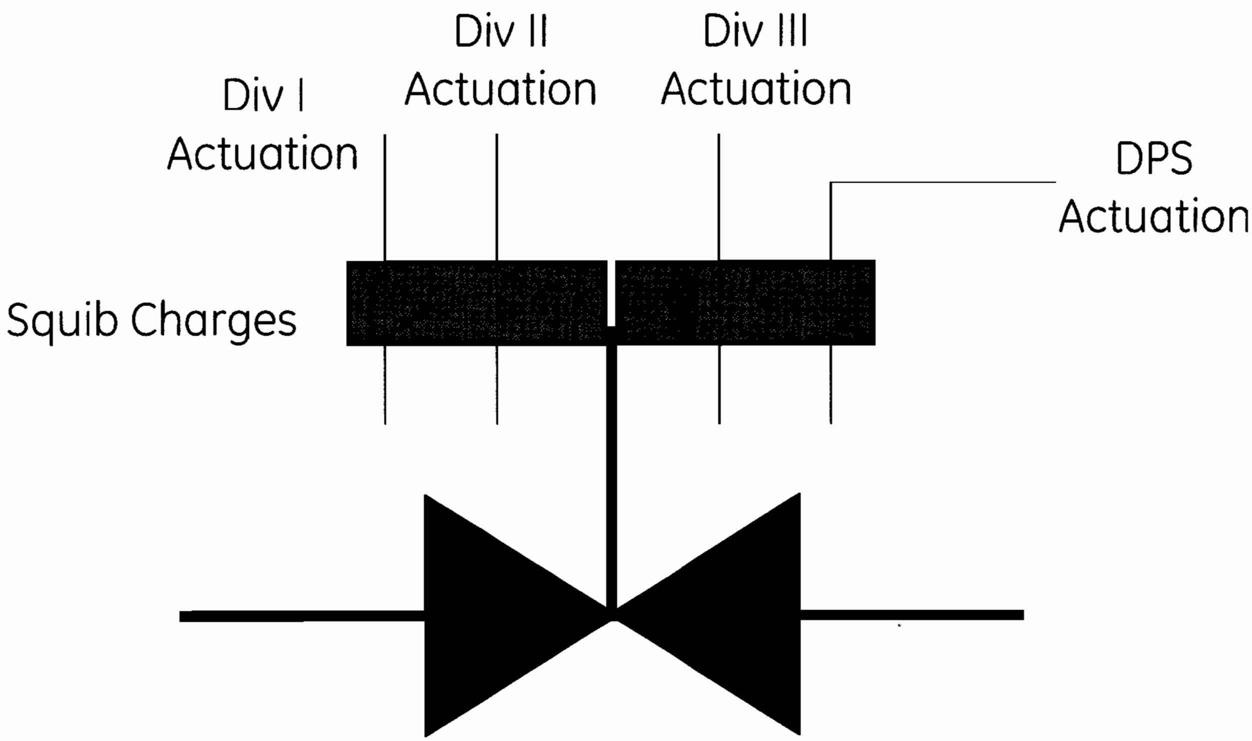
Allows Single Failure Protection with One Division Out of Service

Enables Online Battery Testing

At Least 3 Safety Related Divisions Plus DPS Activates Safety Related Valves

Common Cause is the Only Way to Fail ECCS

Squib Valve Example for N-2



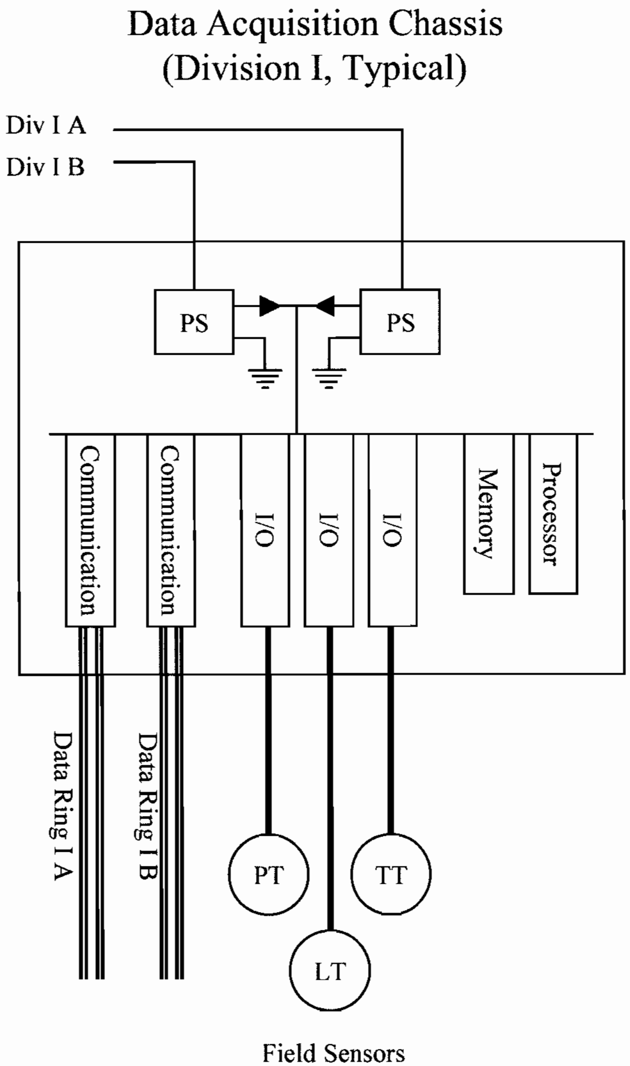
Div I OOS
Div II Fails

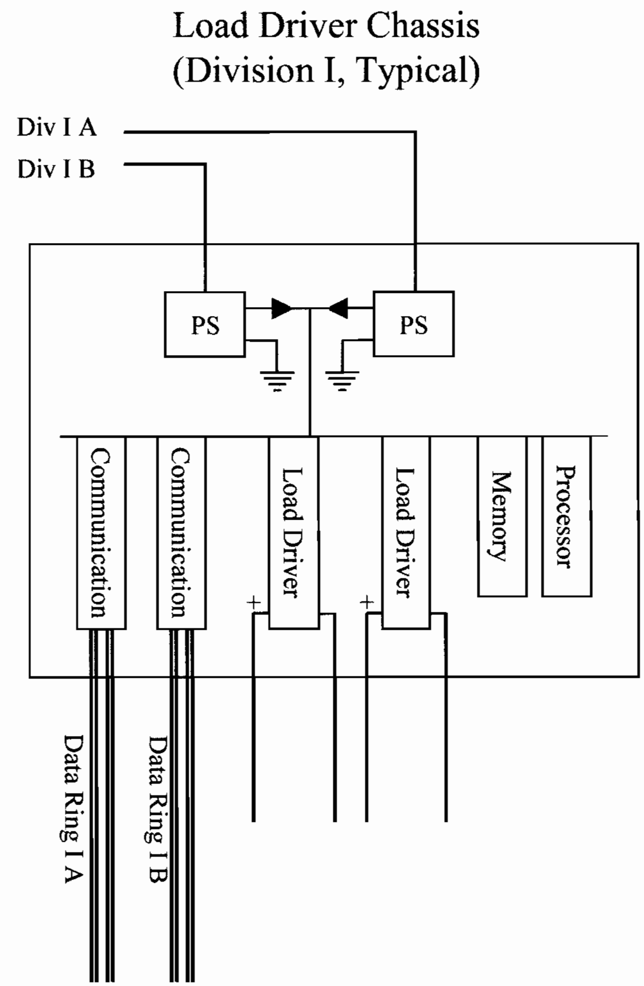
Div III + 4 Provides 2 of 4 Signal
Div III Provides Actuation

Valve Actuation is Successful

DCIS Block Diagram

Next 6 Pages





External Connections

120 V Uninterruptible AC

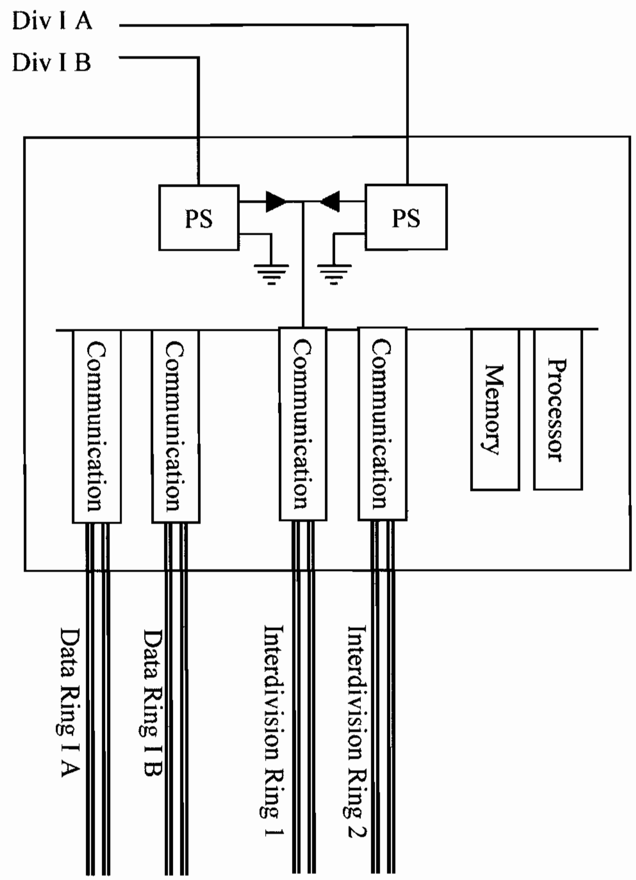
4 – 20 mA Dual Conductor

Dual Counter-Rotating Fiber

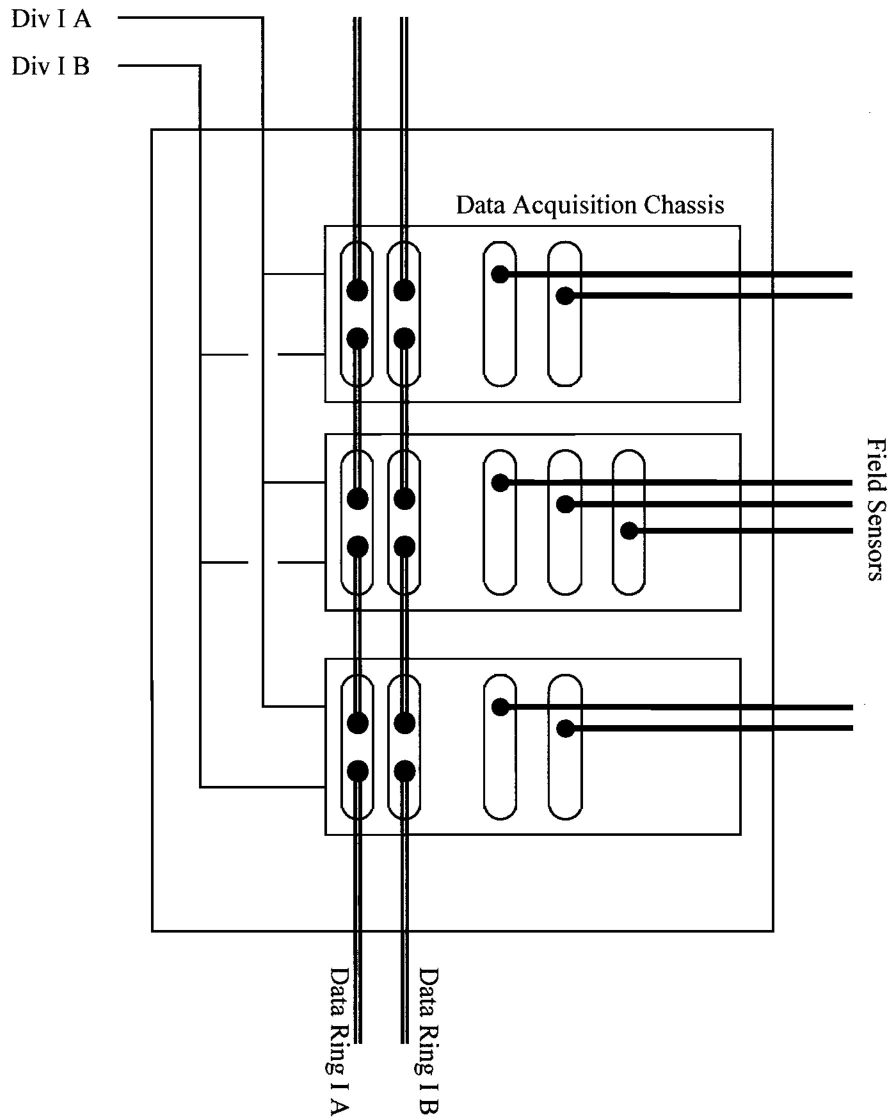
120 V DC Single Conductor

120 V DC Dual Conductor

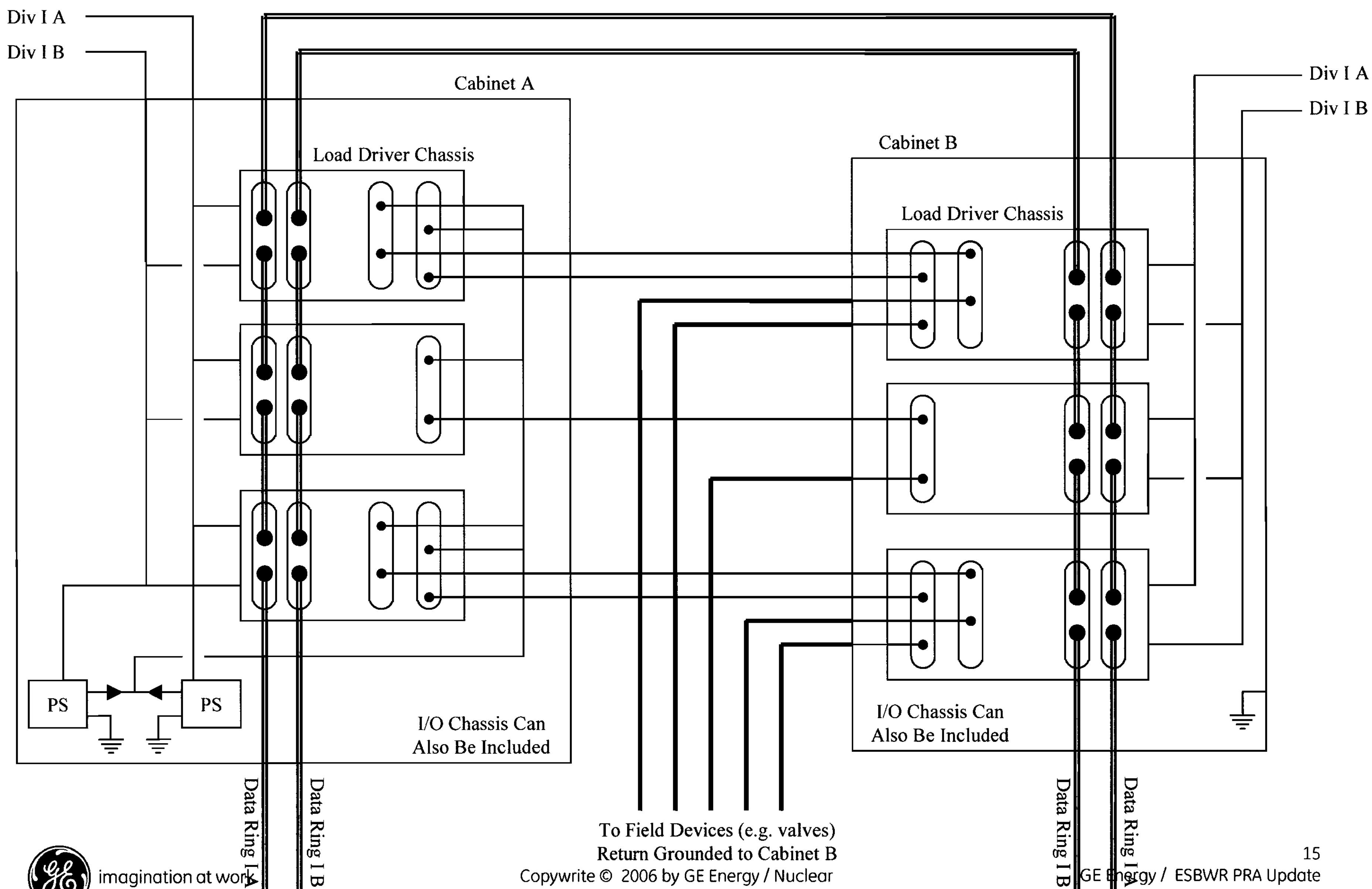
Logic Chassis
(Division I, Typical)

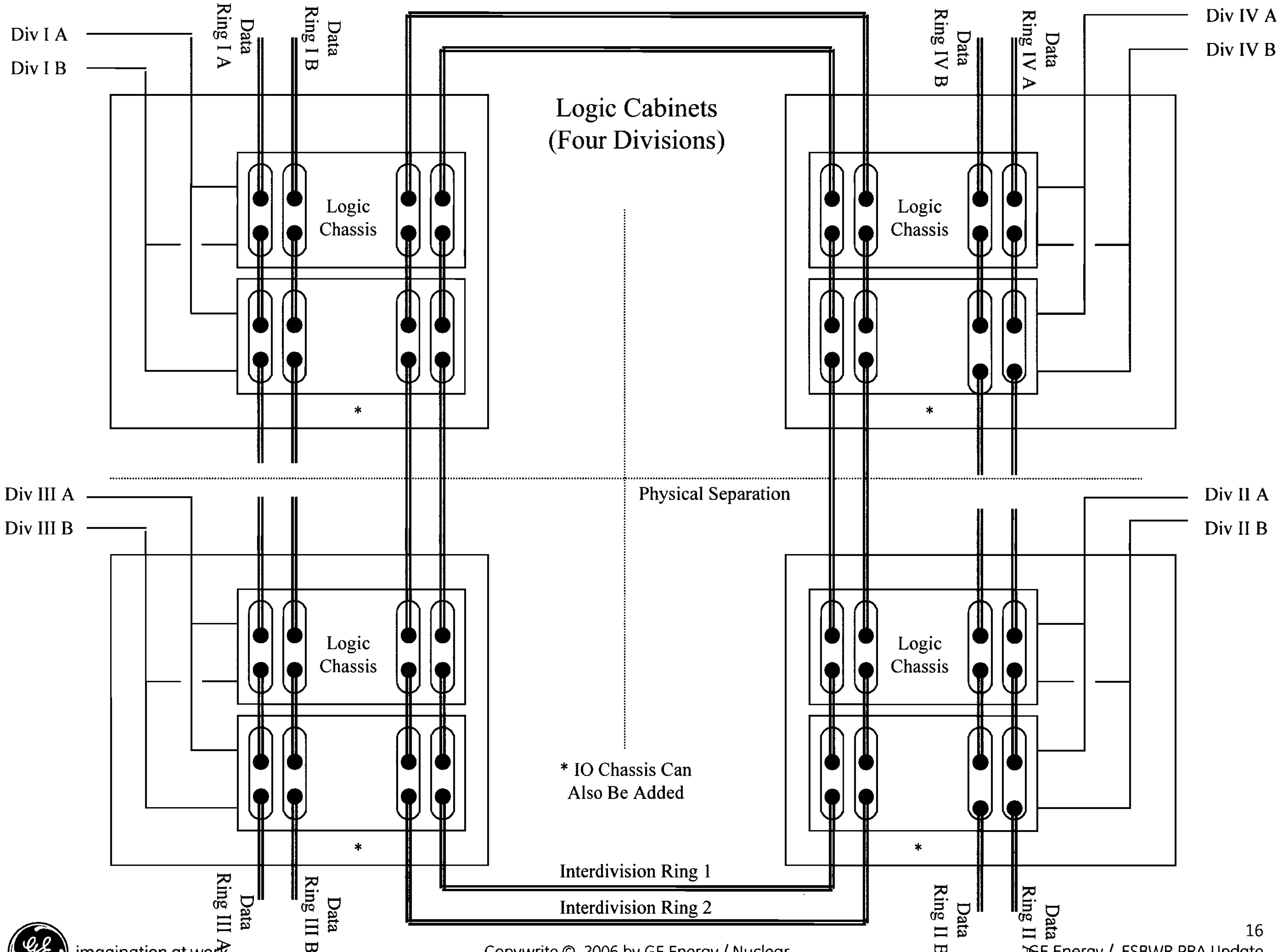


Data Acquisition Cabinet
(Division I, Typical)

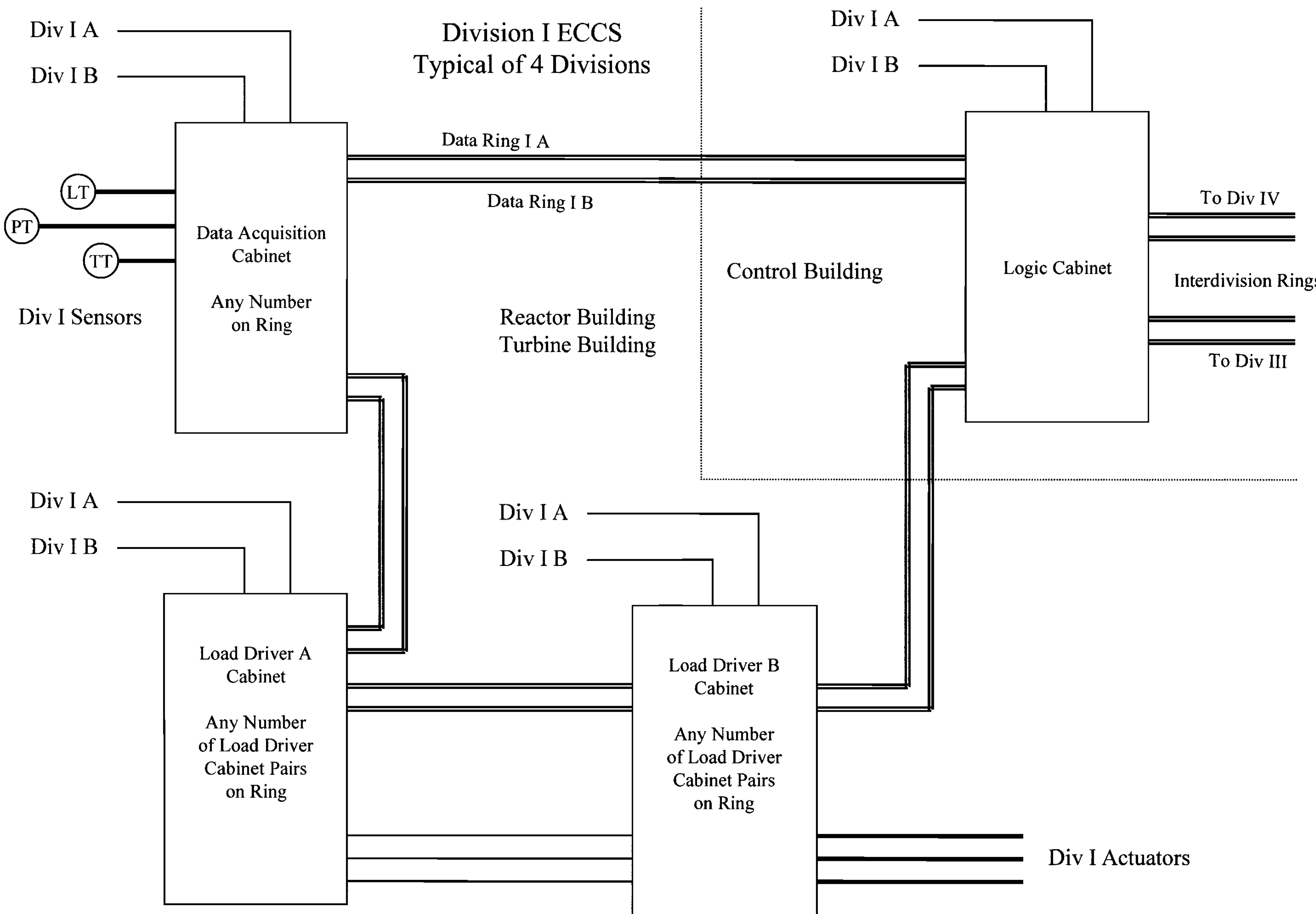


Load Driver Cabinet Pair
(Division I, Typical)





Logic Cabinets
(Four Divisions)
Div I A
Div I B
Data Ring I A
Data Ring I B
Div IV A
Div IV B
Data Ring IV B
Data Ring IV A
Logic Chassis
Logic Chassis
*
*
Physical Separation
Div III A
Div III B
Div II A
Div II B
Logic Chassis
Logic Chassis
* IO Chassis Can
Also Be Added
*
*
Interdivision Ring 1
Interdivision Ring 2
Data Ring III A
Data Ring III B
Data Ring II B
Data Ring II A


Division I ECCS
Typical of 4 Divisions
Div I A
Div I B
Data Acquisition Cabinet
Any Number on Ring
LT
PT
TT
Div I Sensors
Data Ring I A
Data Ring I B
Reactor Building
Turbine Building
Control Building
Div I A
Div I B
Logic Cabinet
To Div IV
Interdivision Rings
To Div III
Div I A
Div I B
Load Driver A Cabinet
Any Number of Load Driver Cabinet Pairs on Ring
Div I A
Div I B
Load Driver B Cabinet
Any Number of Load Driver Cabinet Pairs on Ring
Div I Actuators

Affect of Revision 2 Changes

Top Sequences and Cutsets are All Affected

Adds At Least One Failure Mode to Each Cutset

DCIS Design Provides Additional Protection

Revised Common Cause May Offset

Level 1 Model Results Available in April

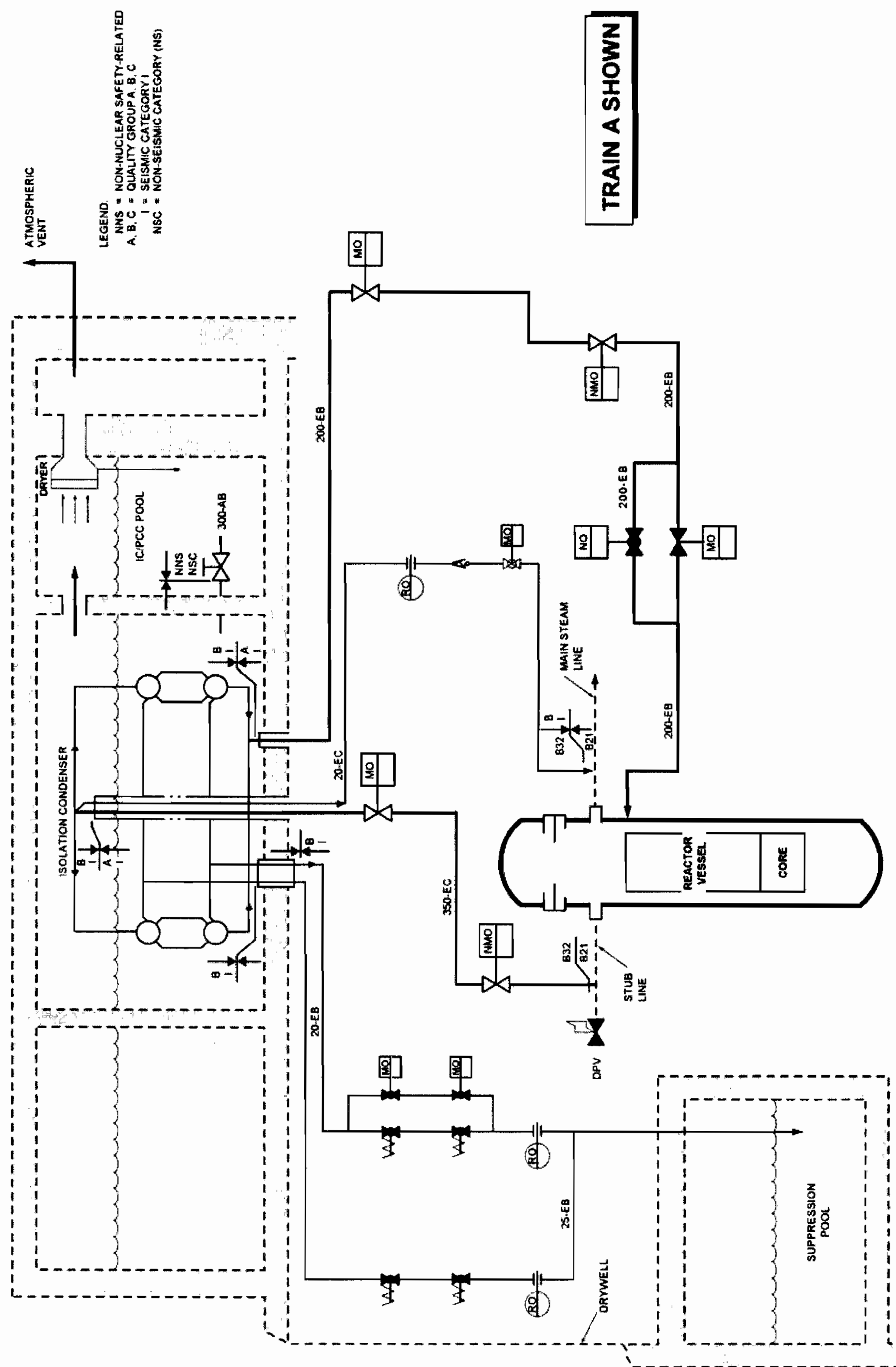


Figure 5.1-3. Isolation Condenser System Schematic

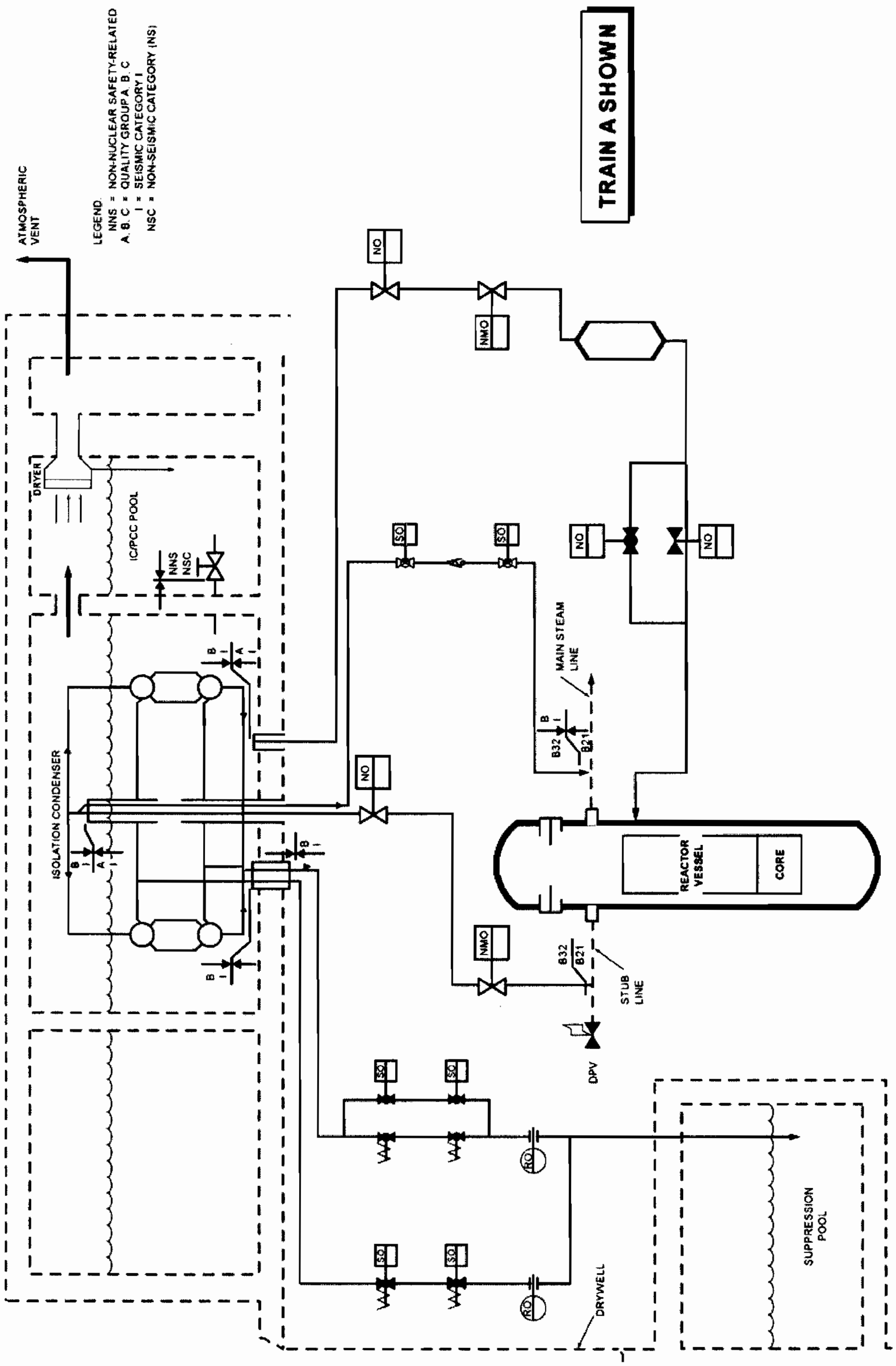


Figure 5.1-3. Isolation Condenser System Schematic

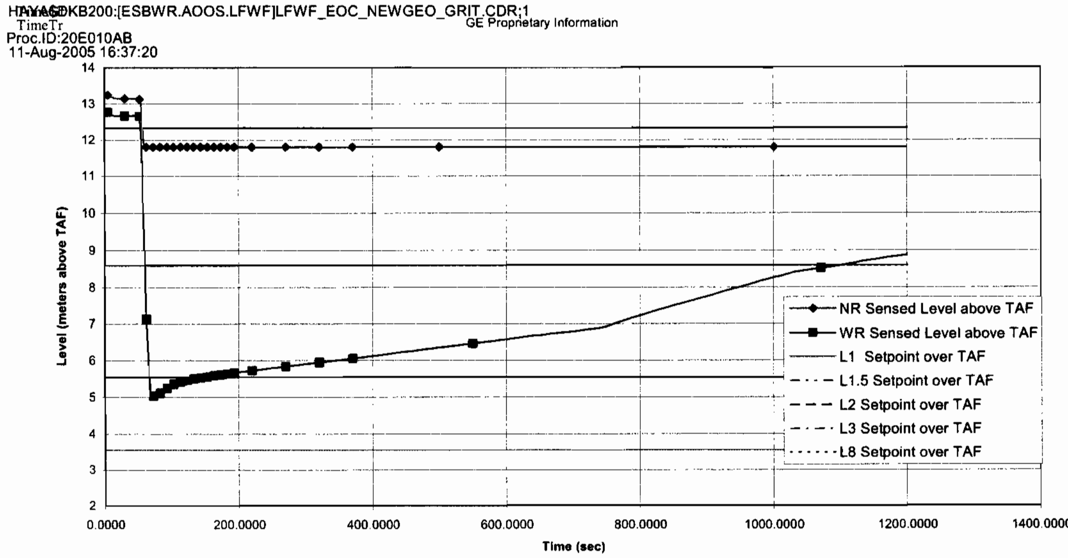


Figure 15.2-16c. Loss of All Feedwater Flow

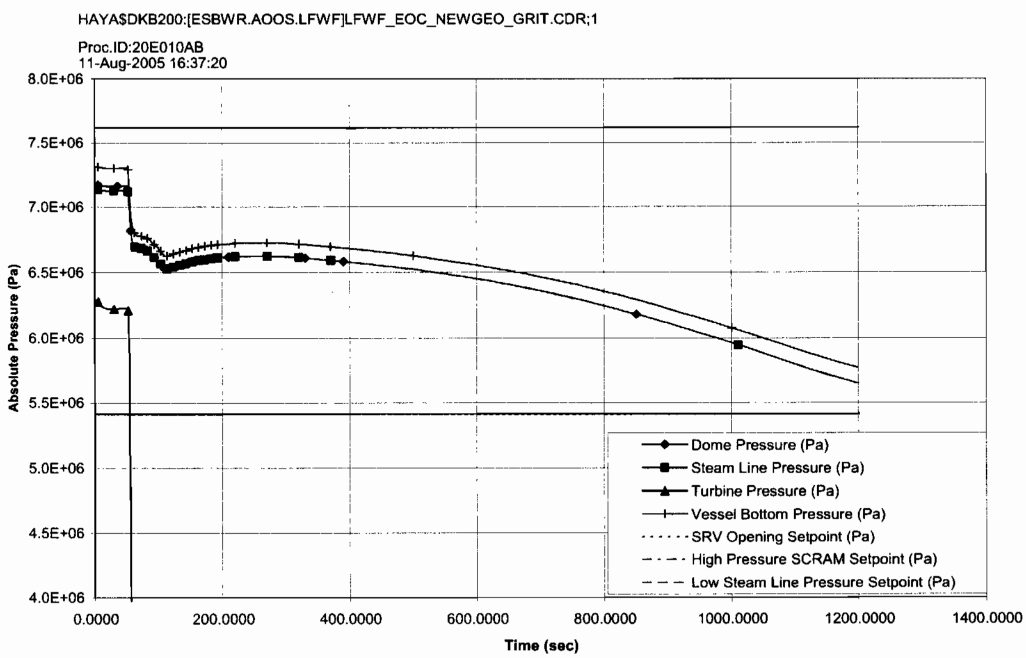


Figure 15.2-16d. Loss of All Feedwater Flow

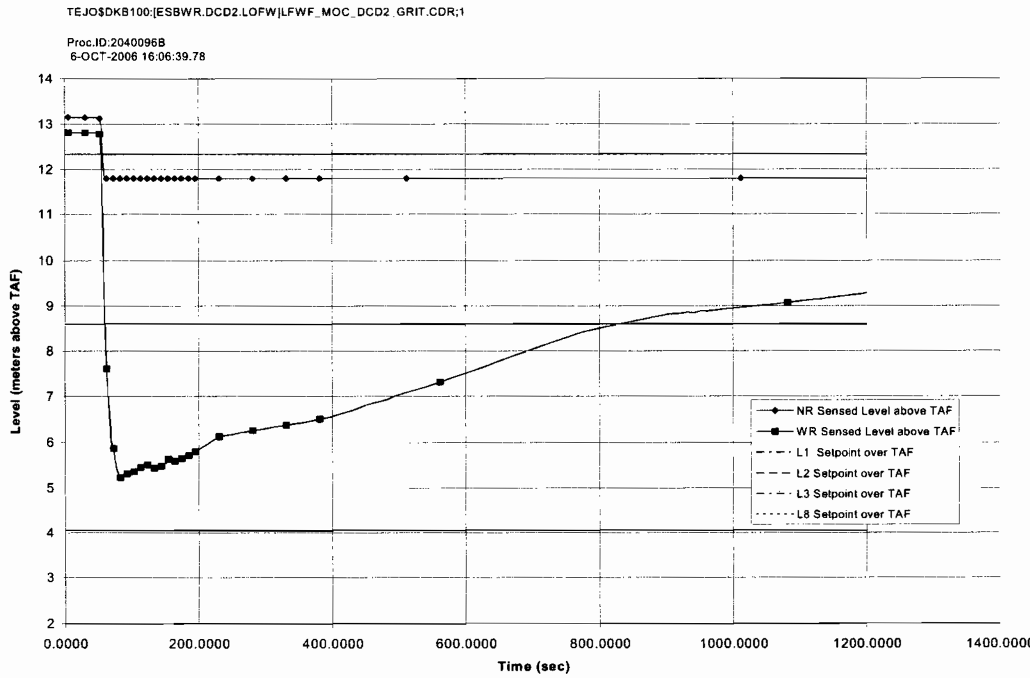


Figure 15.2-16c. Loss of All Feedwater Flow

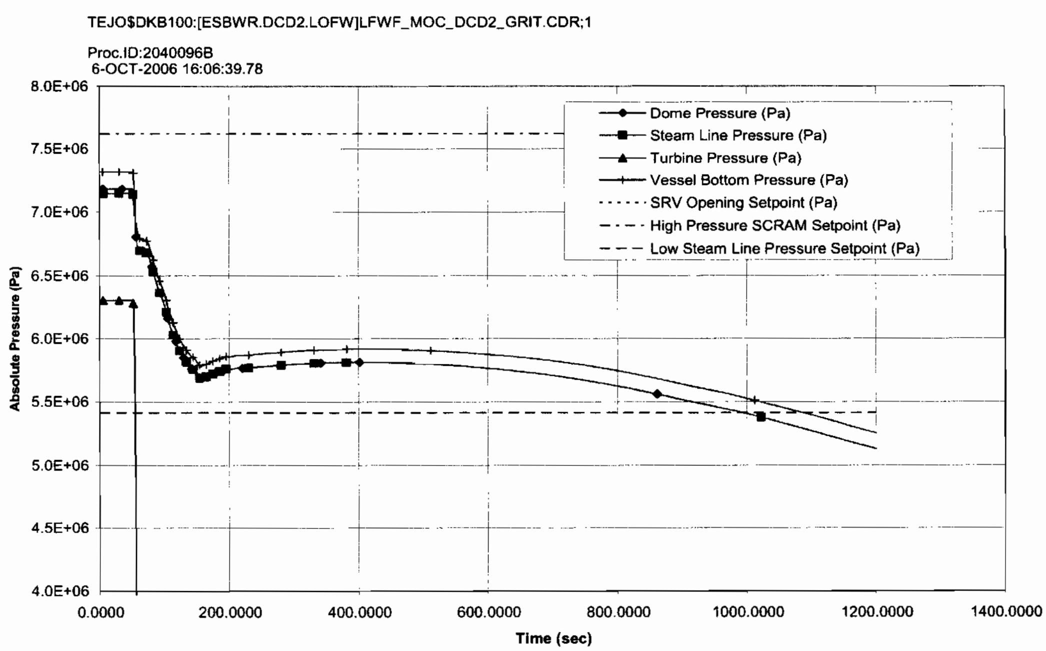
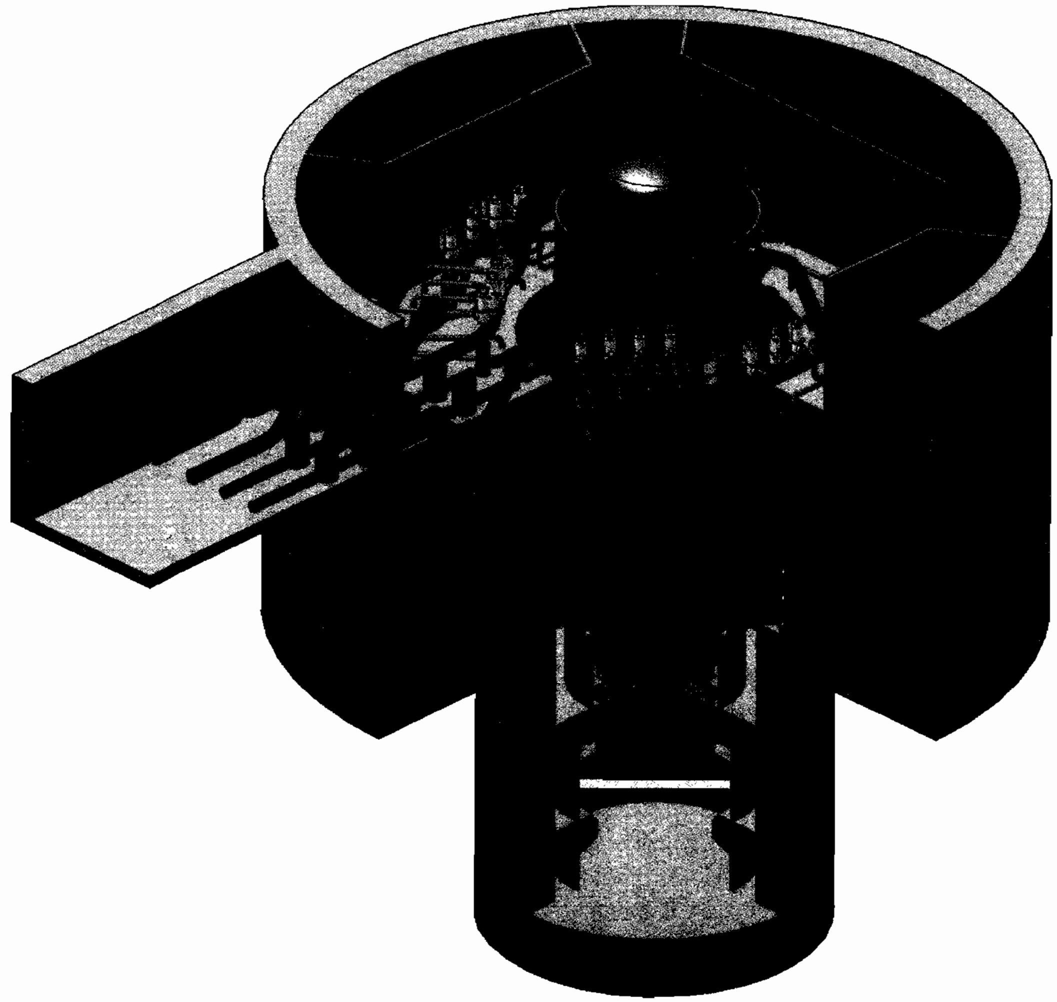


Figure 15.2-16d. Loss of All Feedwater Flow

Probabilistic Risk Assessment

Modeling Issues



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December 14, 2006

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GE Presentation Topics

Common Cause Failure Methods
Data Treatment for Components with Long Test Intervals
Thermal-Hydraulic Uncertainties

Common Cause Model

PRA Rev 1 Used Alpha Factor Method

This Causes Difficulty With:

> Uncertainty Analysis

> Some Sensitivity Analyses

Revision 2 Will Use Multiple Greek Letter Method

> Limit Order of Calculation to β, γ, and δ

Data Sources Under Review

Common Cause Failure Method

The latest release of CAFTA includes a common cause tool.

It enforces a standard method of applying common cause.

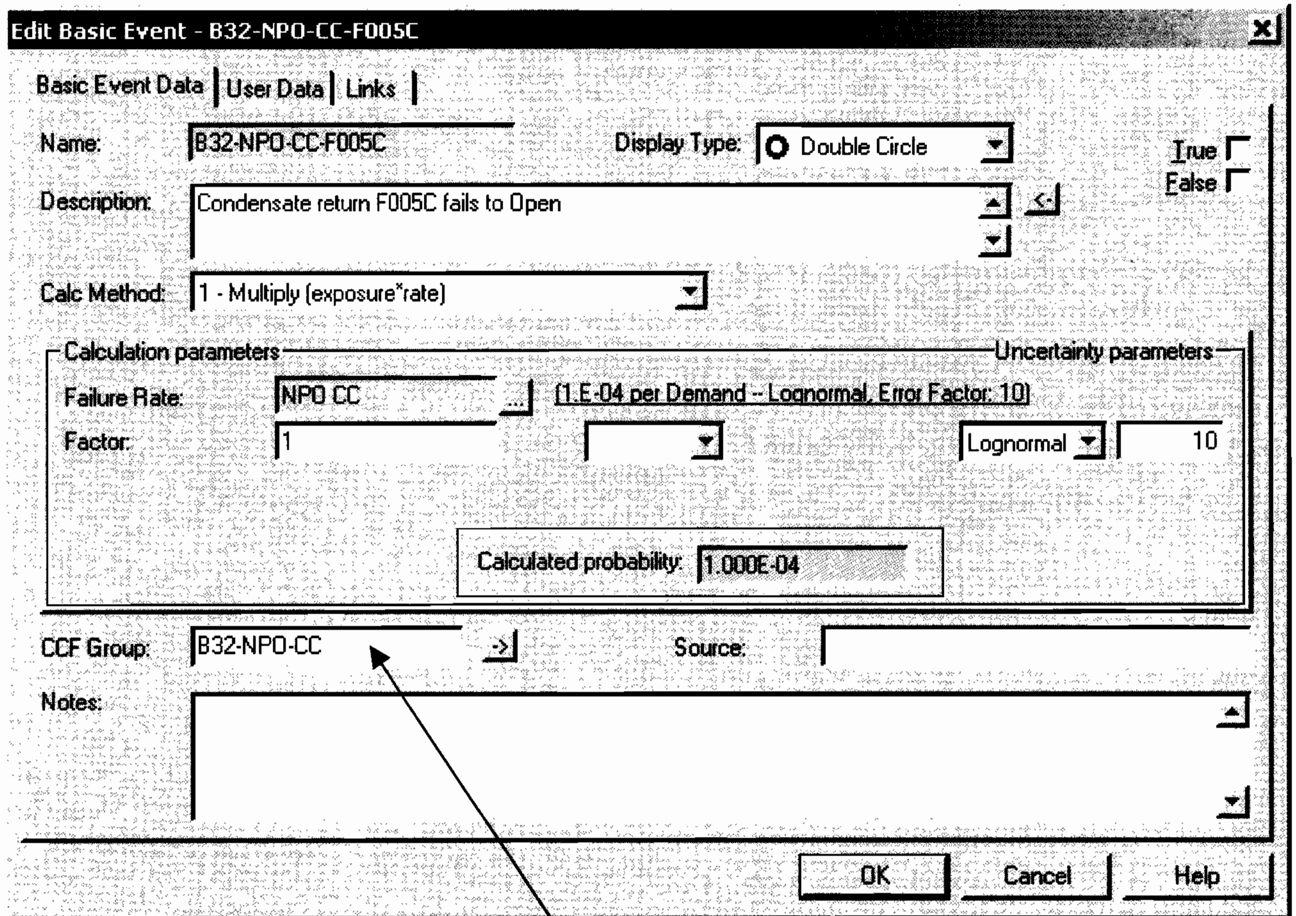
CCF Tool Capabilities

Define CCF Groups

Define CCF Parameters

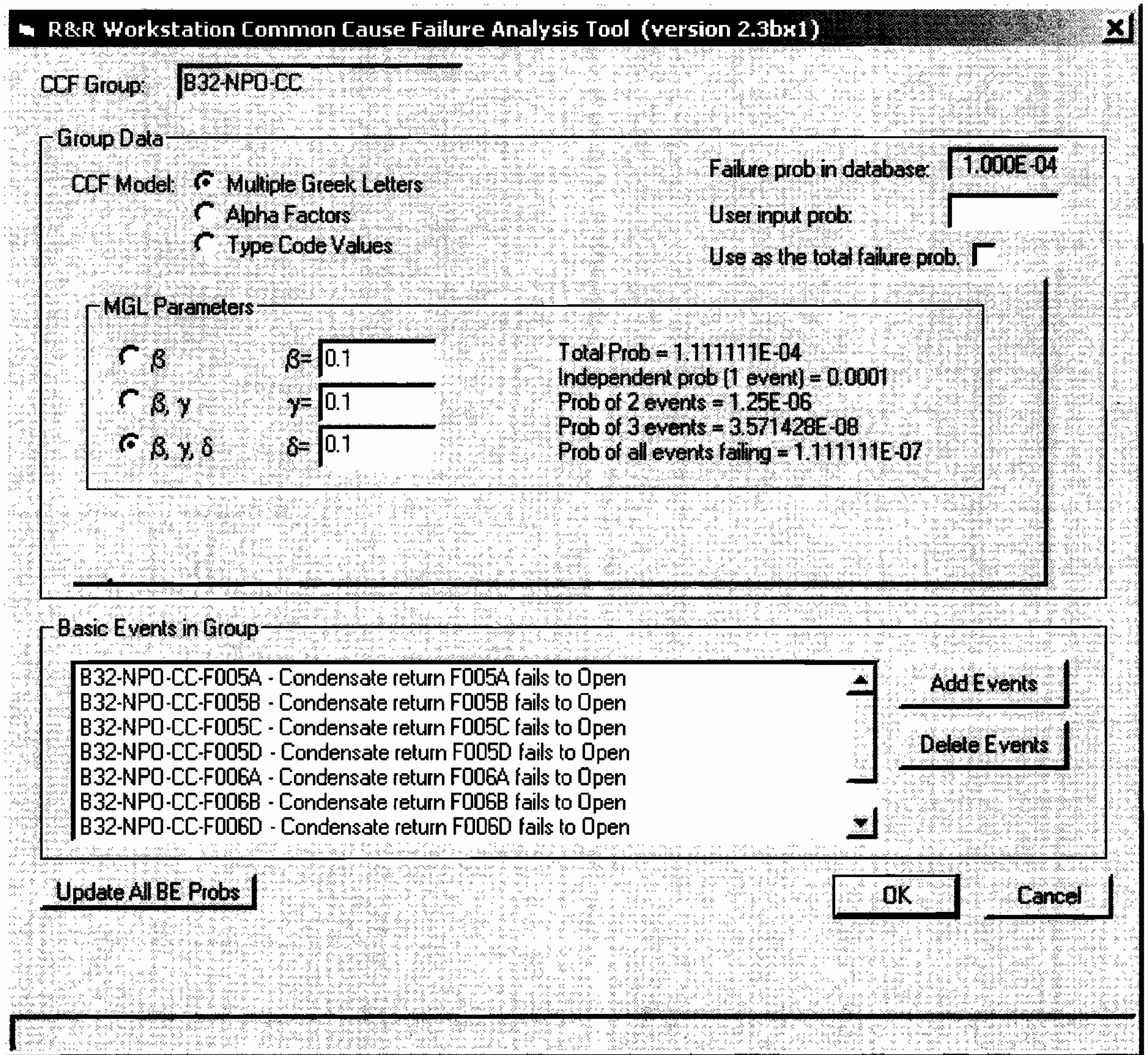
Create CCF Logic

Remove CCF Logic

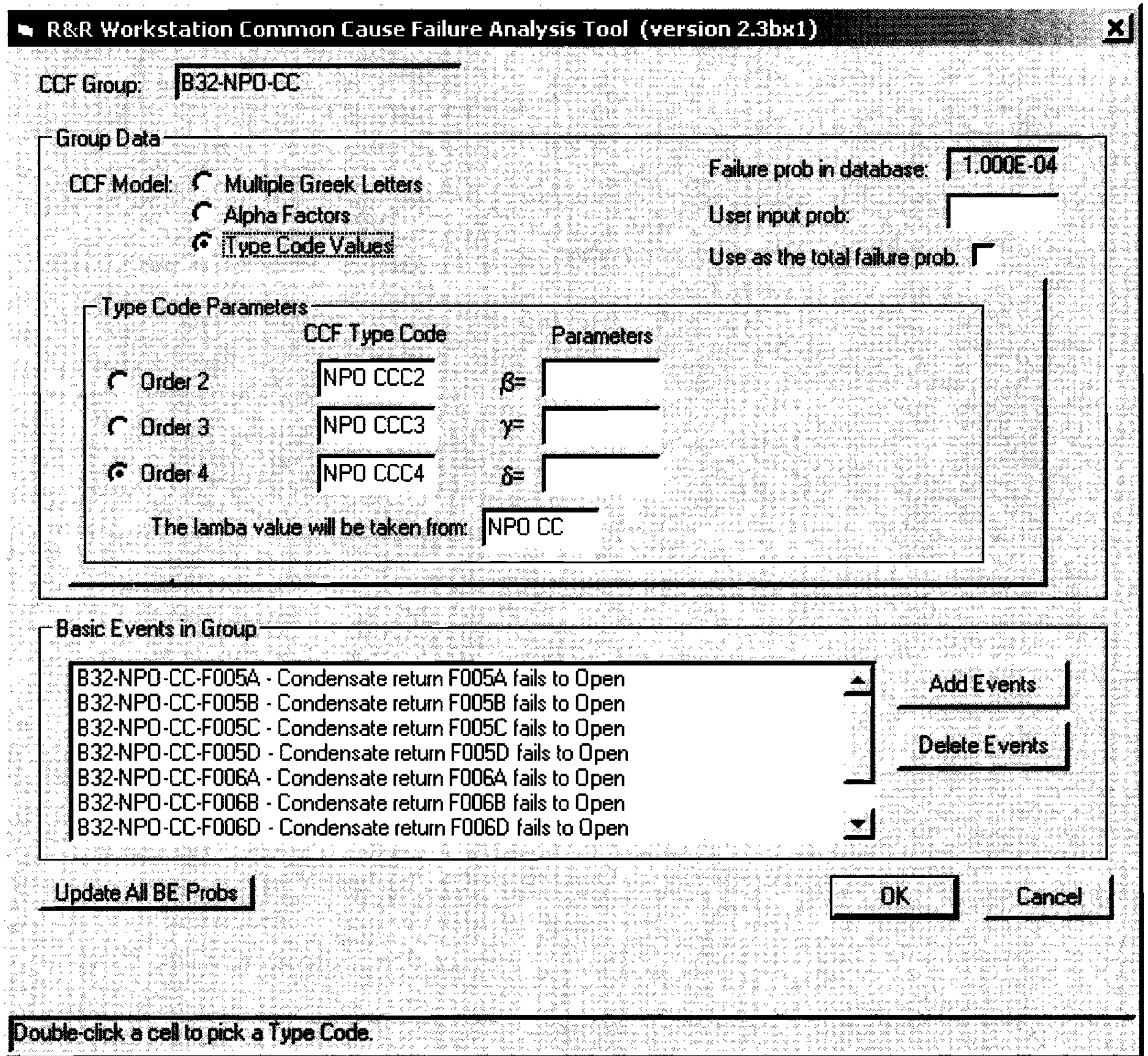


Define the CCF Group

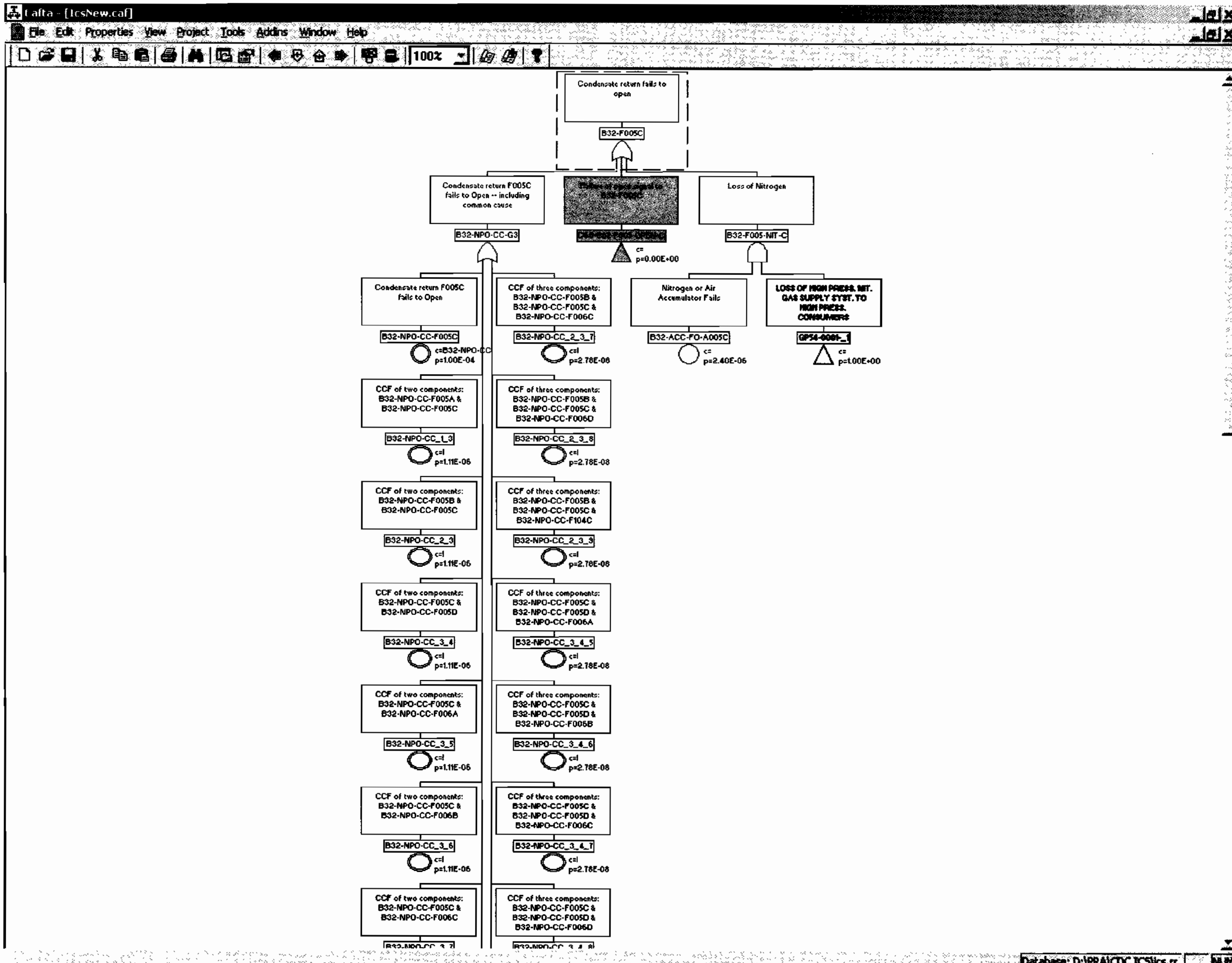
Define the CCF Group Parameters



Type code model



CAFTA does the tedious logic expansion



Cutsets include CCF Terms

Cafta - [Cutset Display]

File Edit Properties View Project Tools Addins Window Help

Disp B32-2LOOPSFAIL 8.3307E-7 Filter: Apply + - Report Compare

1.000E-09	1.000E-05	**B32CNMO_CCNMO-CC_1_2** -
.12%	1.000E-04	**B32-NPO-CC-F104C** - F104C fails to open
1.000E-09	1.000E-05	**B32CNMO_CCNMO-CC_1_2** -
.12%	1.000E-04	**B32-NPO-CC-F104D** - F104D Fails to Open
4.348E-10	1.000E+00	**B32-NONCONDENSE** - Non condensable gasses form in ICS sufficiently to require venting
.05%	1.000E-04	**B32-BV_-MC-F101A** - Manual valve F101A mispositioned closed
	4.348E-06	**B32-SOV-FE_10_18** - CCF of two components: B32-SOV-FE-F009B & B32-SOV-FE-F011B
4.348E-10	1.000E+00	**B32-NONCONDENSE** - Non condensable gasses form in ICS sufficiently to require venting
.05%	1.000E-04	**B32-BV_-MC-F101A** - Manual valve F101A mispositioned closed
	4.348E-06	**B32-SOV-FE_10_22** - CCF of two components: B32-SOV-FE-F009B & B32-SOV-FE-F012B
4.348E-10	1.000E+00	**B32-NONCONDENSE** - Non condensable gasses form in ICS sufficiently to require venting
.05%	1.000E-04	**B32-BV_-MC-F101A** - Manual valve F101A mispositioned closed
	4.348E-06	**B32-SOV-FE_11_19** - CCF of two components: B32-SOV-FE-F009C & B32-SOV-FE-F011C
4.348E-10	1.000E+00	**B32-NONCONDENSE** - Non condensable gasses form in ICS sufficiently to require venting
.05%	1.000E-04	**B32-BV_-MC-F101A** - Manual valve F101A mispositioned closed
	4.348E-06	**B32-SOV-FE_11_23** - CCF of two components: B32-SOV-FE-F009C & B32-SOV-FE-F012C
4.348E-10	1.000E+00	**B32-NONCONDENSE** - Non condensable gasses form in ICS sufficiently to require venting
.05%	1.000E-04	**B32-BV_-MC-F101A** - Manual valve F101A mispositioned closed
	4.348E-06	**B32-SOV-FE_14_18** - CCF of two components: B32-SOV-FE-F010B & B32-SOV-FE-F011B
4.348E-10	1.000E+00	**B32-NONCONDENSE** - Non condensable gasses form in ICS sufficiently to require venting
.05%	1.000E-04	**B32-BV_-MC-F101A** - Manual valve F101A mispositioned closed
	4.348E-06	**B32-SOV-FE_14_22** - CCF of two components: B32-SOV-FE-F010B & B32-SOV-FE-F012B
4.348E-10	1.000E+00	**B32-NONCONDENSE** - Non condensable gasses form in ICS sufficiently to require venting
.05%	1.000E-04	**B32-BV_-MC-F101A** - Manual valve F101A mispositioned closed
	4.348E-06	**B32-SOV-FE_15_19** - CCF of two components: B32-SOV-FE-F010C & B32-SOV-FE-F011C
4.348E-10	1.000E+00	**B32-NONCONDENSE** - Non condensable gasses form in ICS sufficiently to require venting

Cutset#128 Prob: 1.11E-10 Old Prob: 1.11E-10 Generated Database: D:\PRA\CDC ICS\ics.rr NUM

Demand Failure Rates for Equipment with Long Test Intervals

Most Demand Data is Associated with Equipment Tested Quarterly

Some ESBWR Equipment Tested at Much Longer Intervals

Three Methods Proposed, Two Were Used

Demand Failure Rate - continued

Three Cases

1 Test Interval 6 Months or Less

Directly Use Generic Failure Probability

2 Test Interval 6 Months – 1 Year

Use 95th Percentile of Generic Failure Prob

No Components in this Category

3 Test Interval Greater than 1 Year

Convert Demand Failure to Rate (Quarterly Test)

Calculate Unavailability Based on Rate, Test Interval, and No Repair

Demand Failure Rate – Moving Forward

Continue to Use Methods 1 and 3 Only

Re-Evaluate Generic Data for Underlying Test Interval

Thermal-Hydraulic Uncertainty

PRA Success Criteria Is Considered Bounding

Very Few Cases Involve Uncovering Any Fuel

In Those Few, No Significant Heatup Calculated

Concern Remains for Calculating Core Uncovery for Various Sequences

T-H Uncertainty – Original Plan

- Intention Was to Perform Comprehensive Benchmark Between MAAP and TRACG for PRA Sequences
- Demonstrate Accuracy of Predictions by MAAP for Beyond Design Basis Sequences
- Other Priorities Continue to Delay this Activity
- No Probabilistic Efforts Involved

T-H Uncertainty – Current Plan

Minimize the Reliance on Additional TRACG Cases

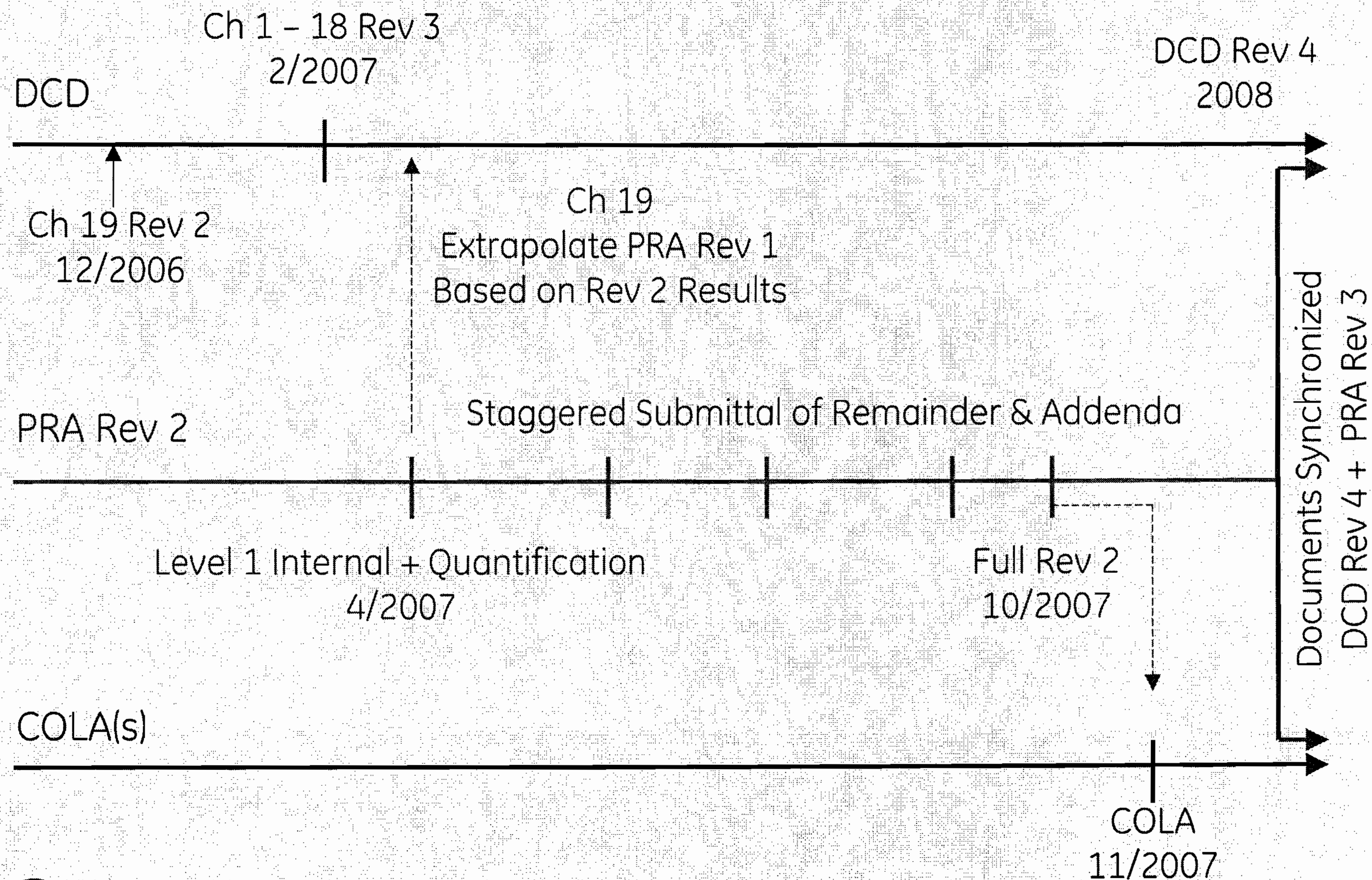
Perform a Quantification of the PRA using Design Basis Success Criteria for Passive Systems

Determine the CDF and LRF Effect

Provide Additional T-H Modeling Only for Sequences that are Outliers

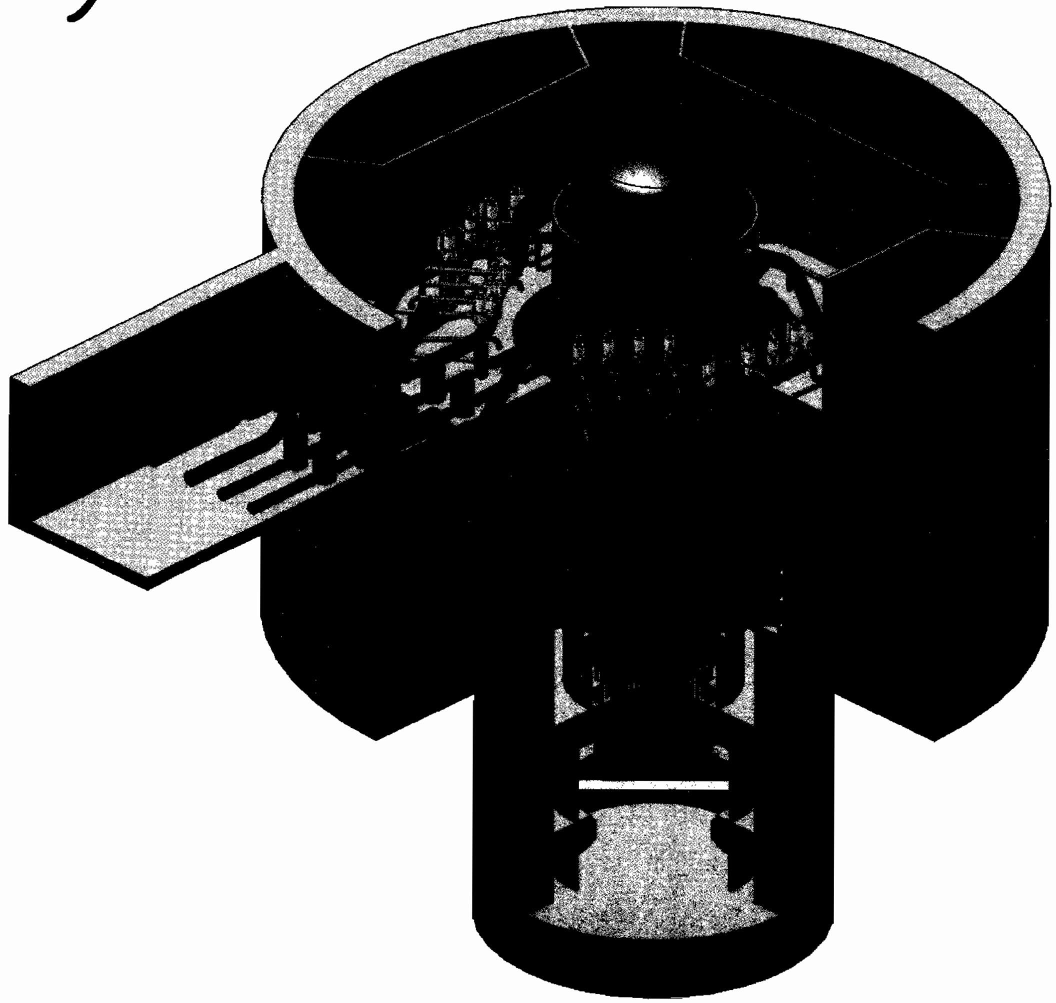
Requires Revision 2 of PRA Model

Certification PRA Timeline



Regulatory Treatment Of Non-Safety Systems

Strategy for the ESBWR Standard Design



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December 14, 2006

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GE Presentation Agenda

Overview of ESBWR RTNSS

> Our understanding of the issue

> Methodology for determining equipment set

> Treatment

Current ESBWR RTNSS Equipment Set

Regulatory Treatment of Non-Safety Systems
- Requirements -

Has Been Required Only for Passive LWR Designs

Regulatory Guidance Contained In:

> SECY-94-084

> SECY-95-132

> Associated SRM's

> Precedent

Deterministic Equipment Selection

Probabilistic Equipment Selection

Regulatory Treatment of Non-Safety Systems
- Equipment Selection Requirements -

Functions Needed to Address ATWS (10 CFR 50.62)

Functions Needed to Address SBO (10 CFR 50.63)

Functions Needed for Post 72 Hour Safety

Functions Needed for Seismic Events

Functions Needed to Prevent Significant Adverse Systems Interactions

Functions Needed to Meet the Probabilistic Safety Goals

ATWS Mitigation - 10 CFR 50.62

Functions Required:

(c)(3) Each boiling water reactor must have an alternate rod injection (ARI) system that is diverse (from the reactor trip system) from sensor output to the final actuation device ....

(c)(4) Each boiling water reactor must have a standby liquid control system (SLCS) with the capability of injecting into the reactor pressure vessel a borated water solution ....

ARI is Non-Safety in ESBWR

SLCS is Safety-Related in ESBWR

Success Using SLCS Requires Successful Feedwater Runback

ARI is RTNSS

Feedwater Controller is RTNSS

Station Blackout – 10 CFR 50.63

ESBWR Has a 72 Hour Coping Period

Nothing More Should Be Required

SECY-94-084

> Diesels or offsite AC power connection can be RTNSS based on other RTNSS criteria

Seismic

Seismic Response Provided By Safety Related Components

> Including seismic margins analysis

Only Issue is Post 72 Hour Safety Following Seismic Event

Long Term Safety

All Initiating Events Are Considered

Required Functions

> > Core Cooling
> > Decay Heat Removal
> > Post Accident Monitoring
> > Control Room Habitability

There Must Be A Strategy For All Contingencies

PRA Used to Determine Risk Significance

Long Term Safety - Phases

0 – 72 Hours	Safety Related, No Operators
3 – 7 Days	Resources Must Be On Site
7 + Days	Off Site Commodity Replacement

More Time Until Needed Results In Less Stringent Requirements

Repair Is OK If Backup Is Available (3 + Days)

All Required Functions Must Be Sustained

RTNSS Based On PRA Results

Systems Needed to Meet Safety Goals

- CDF $\leq 10^{-4}$
- LRF $\leq 10^{-6}$ (and containment performance goal)
- These may be risk significant systems
- Simple Technical Specification treatment

Systems Needed to Address Uncertainty

- These are not risk significant systems
- Maintenance Rule treatment

RTNSS Based on Initiating Events

Three Conditions Must Be Satisfied

> Does non-safety system failure cause initiator?

> Is that initiator risk significant?

- – Contributes approximately 10% to CDF

> Can availability controls reduce initiator frequency?

RTNSS Based on Adverse Systems Interactions

Systematic Approach Used

Failure of Non-Safety Systems That Affect Safety Systems

Actuation of Non-Safety Systems That Affect Safety Systems

Detailed Design Expected to Eliminate All Interactions

Proposed RTNSS Functions

ARI and Feedwater Runback for ATWS

IC/PCC Pool Makeup Via Fire Water

> Diesel pump for 3 – 7 days
> External connection for 7 + Days

Parts of Diverse Protection System

Post Accident Monitoring

> Detailed list in development
> Based on RG 1.97

Additional RTNSS Functions to Address Uncertainty

BiMAC Device

Some Functions of FAPCS

Treatment - Background

DG-1145 Meetings

> Risk significant SSCs
  - Tier 1 & 2 has same level of detail as safety-related
  - ITAAC

> Non-Risk significant SSCs
  - Described in Tier 2
  - Listed in Tier 1

10 CFR 50.36, SECY-93-087, and Precedent (AP1000 FSER)

> Risk significant SSCs need simple technical specifications

> All others have TRM specifications

> All RTNSS included in D-RAP

Treatment - Background

continued

Commercial Grade QA

Maintenance Rule Monitoring for Reliability & Availability

AP1000 FSER

> "Post 72 hour only" functions inherently not risk significant

> Post 72 hour functions are

- Seismic Cat II
- Cat 5 Hurricane Missiles
- Flood Protected

Treatment – Proposed

Functions and Equipment Described in Tier 2

Functions and Equipment Listed in Tier 1

ITAAC for RTNSS Functions Consistent with Significance

Included in D-RAP and Maintenance Rule with Reliability and Availability Targets

Availability Controlled via TRM

QA Meets SRP 17.5.Y (Commercial Grade High Quality)

Post 72 Hour Capability Functions and Equipment:

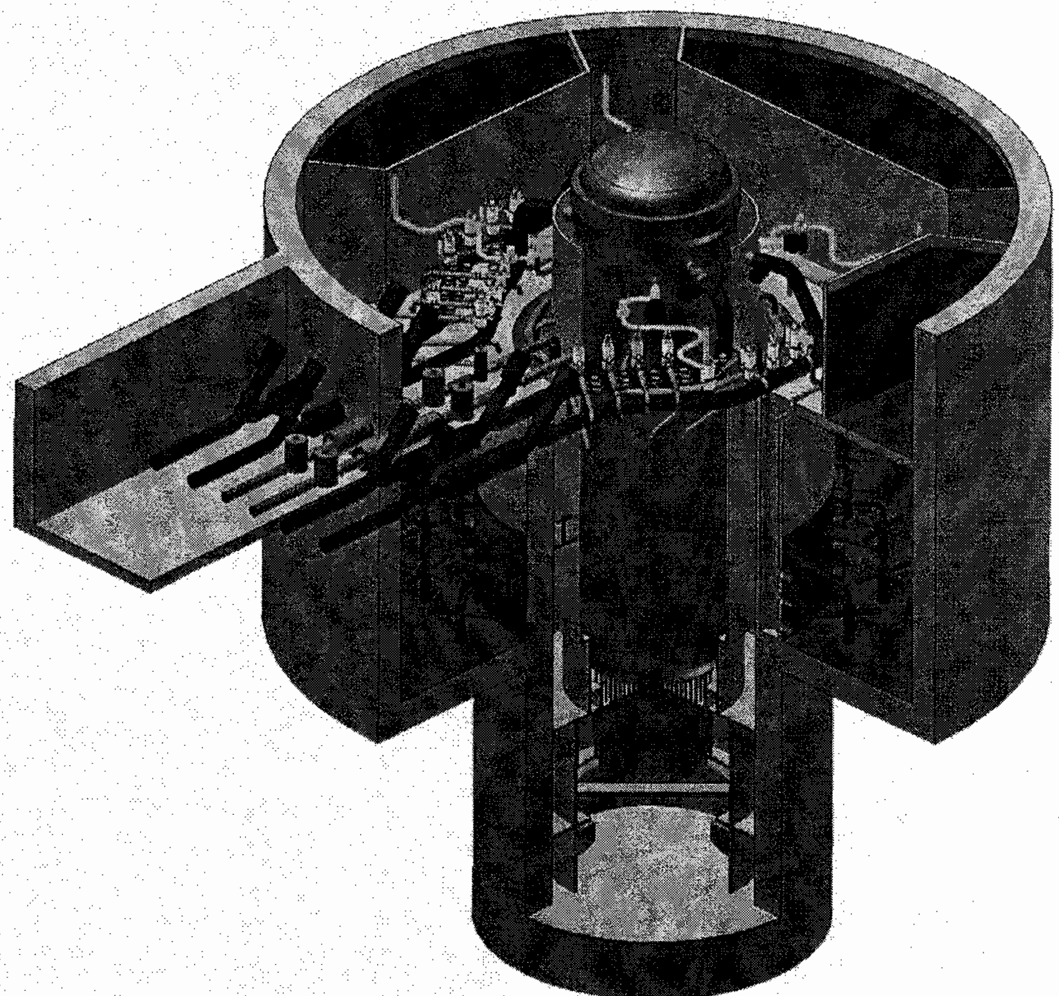
> International Building Code Seismic Standards
> ASCE/SEI 43-05 "Seismic Design Criteria for SSC's in Nuclear Facilities." SDB-5A for Selected Important SSCs
> Withstand Hurricane Cat 5 Missiles
> Protected from Floods

In Addition, Risk Significant Functions and Equipment:

> Described in Tier I
> Availability Controlled via Simple Technical Specifications

Probabilistic Risk Assessment

External Events
Summary



Presented By:
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December 14, 2006

Flood Assessment - Model

Uses Internal Events PRA Model

Evaluates for Worst Case Flood for Each Building

No Mitigation of Flood Effects

Historical Flood Frequencies Used

> PRA initiated design change for Control Building allows special case

> Eliminates potential vulnerability

Flood Assessment - Results

See Tables 13-5 and 13-6

One Significant Full Power Sequence

> Circulating Water pipe break in Turbine Building
> Building reconfigured to eliminate in revision 2

One Significant Shutdown Sequence

> CRD line break
> Can be considered for design change
> Impact very low

Flood Assessment - Insights

Engineered Protection (e.g. Flood Doors) Not Required

> Will not significantly reduce flood risk if credited

Floor Drain System Can Be Simplified

> Consideration of normally closed system

Floods Are Not Significant Contributors to Risk for ESBWR

Fire Assessment - Model

Uses Internal Events PRA Model

Evaluates for Worst Case Fire for Each Division

No Mitigation of Fire Effects (Bounding Assumption)

- Entire division is affected by any fire in that division

Propagation Considered

Fire Frequencies Based on FIVE Tables

Initiators in Revision 2 Will Be Reduced Due to Elimination of Many 480 V Cabinets

Fire Assessment - Results

See Tables 12-15 and 12-16

Most Significant Full Power Sequence (~50%)

> Fire in Turbine Building

> Design changes for revision 2 significantly reduce this sequence

Reactor Building Fire Sequences Also Contribute

> Conservative assumptions drive these results

Reactor Building Fires Dominate Shutdown Fire Risk

Fire Assessment - Insights

Assessment Difficult Without Final DCIS and Electrical Configurations

Risk Will Be Lower When Actual Configuration Is Considered

Fire Barrier Control Is Likely to Be Required During Shutdown

> Needs to be confirmed after final location of equipment is established

Fire And Flood – Overall Impact

Based on the Bounding Assessments Performed

Neither Fire nor Flood Should Be Significant Risk Contributors for ESBWR

Table 12-15

Full Power Core Damage Frequency Due to Internal Fires

Fire Area	Scenario[1]	Fire Scenario Frequency (per year)	Core Damage Frequency (per year)
Control Building DCIS Class 1E Electrical Rooms[2]	Division I Fire (FIRE-CONTROL-BUILD-TGEN-DIVI)	2.76E-03	Negligible
	Division II Fire (FIRE-CONTROL-BUILD-TGEN-DIVII)	2.76E-03	1E-13
	Division III Fire (FIRE-CONTROL-BUILD-TGEN-DIVIII)	2.76E-03	Negligible
	Division IV Fire (FIRE-CONTROL-BUILD-TGEN-DIVIV)	2.76E-03	Negligible
Reactor Building Divisional Zones[2]	Division I Fire (FIRE-REACT-BUIL-DIVI)	1.45E-02	7.42E-10
	Division II Fire (FIRE-REACT-BUIL-DIVII)	1.45E-02	7.76E-10
	Division III Fire (FIRE-REACT-BUIL-DIVIII)	1.45E-02	3.61E-10
	Division IV Fire (FIRE-REACT-BUIL-DIVIV)	1.45E-02	3.88E-10
Non-divisional Areas of Electrical Building; Service Water Building; Control Building non-1E DCIS Rooms; and RCCW and IAS Zones in Turbine Building[2]	Division A Fire (FIRE-NON-DIVISIONAL-REDA)	5.57E-02	5.83E-11
	Division B Fire (FIRE-NON-DIVISIONAL-REDB)	5.57E-02	5.83E-11
Turbine Building	Turbine Building Fire (FIRE-TURBINE-BUILD)	6.14E-02	9.69E-09
Fuel Building	Fuel Building Fire (FIRE-FUEL-BUILD)	1.22E-02	Negligible
Control Room	Control Room Fire (FIRE-CONTROL-ROOM-TGEN)	9.42E-03	7.97E-12

Notes:

(1) Parameters shown in parentheses are the fire scenario initiator IDs used in the accident sequence analysis (refer to Tables 12-17 and 12-19).

(2) The CDF results for these scenarios include both single division fire scenarios and multi-divisional fire scenarios (as described in Section 12.5.1 and Table 12-13).

Table 12-16

Shutdown Core Damage Frequencies Due to Internal Fires

Fire Area	Mode(1)	Scenario(2)	Fire Scenario Frequency (per year)	Core Damage Frequency (per year)
Reactor Building Divisional Zones(3)	5	Division I Fire (FIRE-REACT-BUILD-DIVI-M5)	1.12E-03	7.23E-09
		Division II Fire (FIRE-REACT-BUILD-DIVII-M5)	1.12E-03	7.72E-09
		Division III Fire (FIRE-REACT-BUILD-DIVIII-M5)	1.12E-03	3.63E-09
		Division IV Fire (FIRE-REACT-BUILD-DIVIV-M5)	1.12E-03	4.02E-09
	6 Unflooded	Division I Fire (FIRE-REACT-BUILD-DIVI-M6)	2.80E-04	4.53E-11
		Division II Fire (FIRE-REACT-BUILD-DIVII-M6)	2.80E-04	4.53E-11
		Division III Fire (FIRE-REACT-BUILD-DIVIII-M6)	2.80E-04	6.02E-11
		Division IV Fire (FIRE-REACT-BUILD-DIVIV-M6)	2.80E-04	6.02E-11
Non-divisional Areas of Electrical Building; Service Water Building; Control Building non-1E DCIS Rooms; and RCCW and IAS Zones in Turbine Building	5	Division A Fire Propagates to Division B (FIRE-NON-DIVISIONAL-REDA-M5)	2.20E-03	8E-13
		Division B Fire Propagates to Division A (FIRE-NON-DIVISIONAL-REDB-M5)	2.20E-03	8E-13
	6 Unflooded	Division A Fire Propagates to Division B (FIRE-NON-DIVISIONAL-REDA-M6)	5.45E-04	1.92E-10
		Division B Fire Propagates to Division A (FIRE-NON-DIVISIONAL-REDB-M6)	5.54E-04	1.95E-10
Control Room	5	Control Room Fire (FIRE-CONTROL-ROOM-M5)	3.80E-04	2E-13
	6 Unflooded	Control Room Fire (FIRE-CONTROL-ROOM-M6)	9.43E-05	4.56E-11

Notes:

(1) Fire scenarios during Mode 6-Flooded are not explicitly quantified in the accident sequence analysis. Fires cause loss of DHR scenarios, but during Mode 6-Flooded the time to reach RCS boiling is very long. As such, the risk contribution from Mode 6-Flooded fire scenarios is not significant.

(2) Parameters shown in parentheses are the fire scenario initiator IDs used in the accident sequence analysis (refer to Tables 12-18 and 12-20).

(3) The CDF results for these scenarios include both single division fire scenarios and multi-divisional fire scenarios (as described in Section 12.5.2 and Table 12-14).

Table 13-5

CDF Contribution of At-Power Flooding Scenarios

BUILDING	FLOOD SCENARIO	DESCRIPTION[1]	FREQUENCY (per year)	INITIATING EVENT TYPE[2]	DAMAGE	CDF (per year)[3]
Reactor	AP-1	CRDS Pipe Break Outside Containment (FLOOD-RB-CRD-POWER)	3.40E-03	T-GEN	CRDS, FAPCS and RWCU/SDCS	9.74E-14
	AP-2	FPS Pipe Break Outside Containment (FLOOD-RB-FP-POWER)	3.40E-03	T-GEN	FPS, FAPCS and RWCU/SDCS	Truncated[4]
	AP-3	RWCU/SDCS Pipe Break Outside Containment (FLOOD-RB-RWCU-POWER)	3.40E-03	BOC RWCU	FAPCS and RWCU/SDCS	1.19E-12
Fuel	AP-4	FPS Pipe Break in Fuel Building (FLOOD-FB-FP-POWER)	3.40E-03	T-GEN	FPS, FAPCS and RWCU/SDCS	Truncated[4]
Turbine	AP-5	CIRC Pipe Break, Flood below grade elevation (FLOOD-TB-ALL-POWER)	2.80E-02	T-FDW	PCS	3.68E-09
	AP-6	CIRC Pipe Break, Flood above grade elevation (FLOOD-TB-CIRC-POWER)	2.85E-06	T-PSW	PCS, RWCCS	Truncated[4]
Electrical	AP-7	FPS Pipe Break Outside DG Rooms (FLOOD-EB-FP-POWER)	3.40E-03	T-PCS	13.8 kV buses and Batteries A, A1, A2, B, B1, B2, C	Truncated[4]

Table 13-5

CDF Contribution of At-Power Flooding Scenarios

BUILDING	FLOOD SCENARIO	DESCRIPTION(1)	FREQUENCY (per year)	INITIATING EVENT TYPE(2)	DAMAGE	CDF (per year)(3)
	AP-8	Flood in DG Room A (FLOOD-EBGDA-FP-POWER)	3.40E-03	T-GEN	DG(A)	Truncated(4)
	AP-9	Flood in DG Room B (FLOOD-EBGDB-FP-POWER)	3.40E-03	T-GEN	DG(B)	Truncated(4)
TOTAL AT-POWER FLOODING CDF						3.68E-09

Notes:

(1) Parameters shown in parentheses are the at-power flood scenario initiator IDs used in the accident sequence analysis (refer to Tables 13-7 and 13-9).

(2) Identifies the accident sequence structure used in the CDF quantification. Refer to Section 3 for event tree figures.

(3) The quantification is performed at a truncation limit of 1E-14/yr.

(4) No accident sequence cutsets remained above the quantification truncation limit.

Table 13-6

CDF Contribution of Shutdown Flooding Scenarios

BUILDING	FLOOD SCENARIO	DESCRIPTION(1)	MODE	FREQUENCY (per year)	INITIATING EVENT TYPE(2)	DAMAGE	CDF (per year)(3)
Reactor (Outside Containment)	SD-1	CRD pipe break (FLOOD-RB-CRD-PB5)	5	5.80E-04	Loss of RWCU/SDCS	CRDS, FAPCS and RWCU/SDCS	1.32E-13
	SD-2	FPS pipe break (FLOOD-RB-U43-PB5)	5	5.80E-04	Loss of RWCU/SDCS	FPS, FAPCS and RWCU/SDCS	Truncated(4)
	SD-3	CRD pipe break (FLOOD-RB-CRD-PB6)	6-Unflooded	1.44E-04	Loss of RWCU/SDCS	CRDS, FAPCS and RWCU/SDCS	1.49E-09
	SD-4	FPS pipe break (FLOOD-RB-U43-PB6)	6-Unflooded	1.44E-04	Loss of RWCU/SDCS	FPS, FAPCS and RWCU/SDCS	8.69E-11
	SD-5	RWCU/SDC pipe break (%BOC-RWCUSD5)	5	5.80E-04	BOC RWCU	FAPCS and RWCU/SDCS	7.37E-14
	SD-6(5)	RWCU/SDC pipe break (%BOC-RWCUSD6)	6	5.87E-04	BOC RWCU	FAPCS and RWCU/SDCS	1.12E-13
Fuel	SD-7	FPS pipe break (FLOOD-FB-U43-PB5)	5	4.36E-04	Loss of RWCU/SDCS	FPS, FAPCS and RWCU/SDCS	Truncated(4)
	SD-8	FPS pipe break (FLOOD-FB-U43-PB6)	6-Unflooded	1.08E-04	Loss of RWCU/SDCS	FPS, FAPCS and RWCU/SDCS	6.26E-11
TOTAL SHUTDOWN FLOODING CDF							1.64E-09

Notes:

(1) Parameters shown in parentheses are the shutdown flood scenario initiator IDs used in the accident sequence analysis (refer to Tables 13-8 and 13-10).

(2) Identifies the accident sequence structure used in the CDF quantification. Refer to Section 16 for event tree figures.

(3) The quantification is performed at a truncation limit of 1E-14/yr.

(4) No accident sequence cutsets remained above the quantification truncation limit.

(5) Both Scenarios 6a (Mode 6-Unflooded) and 6b (Mode 6-Flooded) are included in this line item summary.

ESBWR
PROBABILISTIC RISK ASSESSMENT

STAFF UPDATE

ACRS - Reliability and Probabilistic Risk Assessment Subcommittee

December 14-15, 2006

Requests for Additional Information (RAIs)

- SER Chapter 19
- RAIs (PRA Rev 1)
  - Issued: 157
  - Review continues
- Responses (complete/partial): 84
- Responses remaining
  - Initial response: 73
  - Supplemental response: 15+
- Effect of PRA Revision 2 – to be determined

Schedule & Resources

Schedule

- ESBWR Design Certification
  - Design Control Document – Chapter 19, Revision 2
  - PRA Revision 2
- ESBWR Combined License Applications
  - Initial Applications – November 2007

Staff Resources

- Coordination of Parallel Reviews
  - Design Control Document
  - PRA (Rev 1 RAIs + Rev 2)
  - COL Applications
- Parallel SERs

Key Technical Review Issues in Level 1 (At Power) PRA

- Common Cause Failure (CCF) Probabilities
- Modeling of I&C Systems
- PRA Mission Time
- Modeling of Steam Suppression Vacuum Breakers
- Fire-Risk Issues
- PRA Input to the Licensing Basis
- Thermal-Hydraulic Uncertainty and Success Criteria

Key Technical Review Issues in Shutdown PRA

- Large Early Release Frequency Risk
- Role of Operator
- Common Cause Failure of Non-Safety Systems
- Risk Impact of Open Containment

Key Technical Review Issues in Level 2 PRA/Severe Accidents

- Basemat-Internal Melt Arrest and Coolability (BiMAC) System
- Vacuum Breakers